FISSION PRODUCT DECAY CHARACTERISTICS

THESIS

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PREFACE

The purpose of this study was to examine fission product decay characteristics and their effects on the Source Normalization Constant. The fission product data was obtained from two different isotope generation and depletion codes and the exposure rate calculations were performed using the method of successive scatters to determine the contribution from scattered radiation.

Several people contributed to the successful completion of this study. Dr. Tal England of the Los Alamos National Laboratory, provided the code DKPOWR, and valuable information on fission pulse data. I would like to thank Dr. Charles J. Bridgman for his guidance and encouragement throughout the study. I would also like to acknowledge the members of the committee, LT CMDR Kirk Mathews and Dr. George John, for their insights and suggestions. Finally, I greatly appreciate the help and tolerance from my wife, Andrea.

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Abstract

The purposes of this study were to determine fission product decay characteristics, including the total activity, the gamma-ray emission rate (GER) and gamma-ray energy spectra. The activity and GER decay were compared to Way and Wigner's $t^{-1.2}$ approximation, and the effects the spectra, activity and GER have on the Source Normalization Constant (K) were examined. Most of the fission product data were obtained from DKPOWR, and were compared with data obtained from ORIGIN2. Since the gamma rays are of primary concern in fallout studies, the GER is used instead of activity. The ratio of GER to activity changes significantly with time. The results of this study calculate a GER of 590×10^{18} gamma rays/second per kT of fission yield from U-235 fuel and a K of 7059 R/Hr/(kT/km^2). The calculation of K includes the contribution from scattered photons. The GER result is 11% higher than reference values, while the K is within 2% of the current value in Glasstone and Dolan's *The Effects of Nuclear Weapons*. The Ks for Pu-239 and U-238 were within 5% of the U-235 results. The Way-Wigner $t^{-1.2}$ approximation differs from time dependent GER and K up to 85% for times less than 6 months. The approximation is not valid for the GER or K at times greater than 6 months. The approximatin is within about 45%, for the activity from fission product decay to atleast 5 years. A more accurate measure of exposure requiree a numerical integration of the time dependent GER and Source Normalization "Constant".

Fission Product Decay Characteristics

I Introduction

Background

The fundamental reason for evaluating fallout patterns is to calculate the exposure rate to people and equipment due to ground and air contamination from radioactive particles. The exposure rate calculations, as currently done, are based on several variables and approximations.

As an example, the exposure rate calculations in Glasstone and Dolan's "The Effects of Nuclear Weapons", a commonly used reference for nuclear weapon effects studies, present no analysis of the energy spectrum emitted by the fission products. Simply an average energy of 700 keV is suggested at one hour following the detonation (10:454). No discussion is offered concerning any difference of the products from various fuels or how the energy may depend on time.

The number of gamma-ray emissions per second (GER) from the products created from 1 kiloton of fission yield is another uncertain term in the equation. Although Glasstone never explicitly defines the term Gamma-Megacurie, he states that 530 Gamma-Megacuries of activity are released from a kiloton of fission yield at one hour following the detonation (10:453). Again, differences among fuels are not addressed. In this text, the term Gamma-Megacurie is defined as

3.7×10^{18} gamma-ray emissions per second. The difference between gamma-ray emission rate and activity (disintigrations/second), is that various isotopes can emit none, one, or several gamma rays from each disintegration. Additionally, the rate at which the activity and gamma-ray emission rate decreases with time is also uncertain. Way and Wigner (15:1318) determined an approximation for the decay rate in 1948, however Glasstone suggests that the approximation is unreliable at times below 30 minutes and greater than 6 months (10:450).

Two isotope generation and depletion codes, ORIGIN2 and DKPOWR, were used in this study to examine the amount of activity released, the GER, the gamma-ray energy spectrum emitted, and the decay of the fission products at early and late times. ORIGIN2 is a reactor code created by Oak Ridge National Laboratory. It was used here with 20 second high power runs to emulate a weapon pulse. DKPOWR, however, is used by Los Alamos National Laboratory and is well suited to fission pulse studies. Both codes were run on the AFIT Scientific Support Computer (SSC), a VAX 11/785 with a UNIX operating system.

Purpose and Scope

The following goals were established for this study:

(1) determine the amount of activity and the gamma-ray emission rate produced from the fission product decay following a nuclear detonation,

(2) examine the nature of the gamma-ray energy spectrum emitted

by the fission products from the three major fuels, U-235, U-238 and Pu-239,

(3) compare the rate of decay of the combined fission products with Way-Wigner's approximation, and

(4) evaluate the effects these factors have on exposure rate calculations.

As with all fallout studies several assumptions and limitations must be noted. (1) No analysis was given to the induced activity from the weapon. Based on the weapon being evaluated this could cause some significant differences in both the amount and characteristics of the activity released. (2) The fission products calculated by DKPOWR for U-235 and Pu-239 fuels are based on thermal fission, while the ORIGIN2 calculations were based on fast reactor fission. Obviously, the U-238 data from both codes is based on fast fission. Neither code has the cross-section or fission yield data to determine the fission products from weapon energy neutrons. T. England, of the Los Alamos National Laboratory uses thermal fission for U-235 and Pu-239 and considers fuel type to be more important than neutron energy when analyzing the fission products and their decay characteristics (7:*). Further assumptions will be noted where appropriate.

II Theory

Fission Product Data

There are two methods available to determine the characteristics of fission product decay. The most accurate and, of course, the most difficult, is to track all the individual isotopes released from the fission for all times of interest. This would include accounting for nearly 1300 initially-produced isotopes (8:1). The second method is to apply statistical models of the fission product generation and decay. K. Way and E. P. Wigner were assigned to perform this task in 1946 and based on available resources chose the statistical approach (15:1318). Both methods will be examined in this study.

Isotope generation and depletion codes are used to determine what fission products were created and which still remain at times of interest. The fission yield is known for each individual isotope; therefore the initial complement of each fission products is:

$$N_i = \beta_i F \tag{1}$$

where

N_i = Number of atoms of isotope i created from fission

β_i = Fission yield for isotope i

F = Total number of fissions

The fission products subsequently undergo a series of decays until reaching a stable isotope. The primary means of decay is by β^-, both

with and without a coincident gamma ray release, although other decays including alpha and neutron emission are possible.

The following non-homogeneous, first-order, ordinary differential equation describes the isotopic concentrations:

$$\frac{dN_i}{dt} = \sum_{j=1}^{J} l_{ij}\lambda_j N_j - \lambda_i N_i \qquad (2)$$

where

$\frac{dN_i}{dt}$ = Time rate of change of the number density of nuclide i

N_i = Atom density of nuclide i

J = Total number of nuclides

l_{ij} = Fraction of radioactive disintegrations by nuclide j that leads to the formation of species i

λ_i = Radioactive decay constant for nuclide i

This equation is valid only after the fissioning event and must be solved for each of the nearly 1300 isotopes created from the fission. If a neutron flux is present, i.e. the fuel is still in a reactor, there will be additional terms in the equation. For each nuclide$_i$ there will be a source term due to continued fissioning, a loss term due to neutron absorption, and a source term from other nuclides which

absorb neutrons and create nuclides.

DKPOWR and ORIGIN2 are computer codes used for reactor studies, however DKPOWR is well suited for pulse work, ORIGIN2 is not. DKPOWR is a code which uses a non-linear least squares fit to results from CINDER-10, an isotope generation depletion code used at Los Alamos National Laboratory (LANL). DKPOWR supplements the CINDER-10 results with experimental data for early-time U-235 and Pu-239 spectra (16:1.1). A description of DKPOWR and its input file is presented in Appendix J. For a detailed description of the ORIGIN2 program and its input file structure see *ORNL/TM-7175* (4:25-61a).

The activity as a function of time is calculated in both codes by summing the contributions from each individual isotope. Glasstone uses the term "Gamma-Megacurie" to describe the gamma-ray activity from the weapon. however the term is not specifically defined. The definition used in this study is a Gamma-Megacurie = 3.7 X 10^{16} gamma-ray emissions per second. Using the terms "activity" and "curies", is misleading; the value of interest in fallout calculations is the gamma-ray emission rate (GER), not the activity. The difference between the GER and the activity is that each time an isotope decays, it may emit none, one or several gamma rays. The relationship between GER and activity is:

$$GER = DPS \cdot GPD \qquad (3)$$

where

DPS = The number of disintigrations per second

GPD = The average number of gamma rays emitted per disintigration

The spectra emitted by the fission products is presented by the ORIGIN2 output in 18 discrete energy bins, with the bin centers defined. The spectra given by DKPOWR output is in 19 bins with the boundary values given. A bin description for DKPOWR output is shown in Appendix J. The midpoint energies of the bins were used for the calculations.

Way and Wigner's commonly used $t^{-1.2}$ approximation for fission product decay was based on a statistical treatment of the radioactive decay process (15:1318). The use of a statistical model is valid when numerous isotopes are involved in the decay process, but later in time the model may begin breaking down as a few isotopes begin to dominate the decay. Since Way and Wigner's approximation was only concerned with disintegrations, the gamma emission rate may not be approximated by the same time dependence. As noted above, the value of interest is gamma-ray emission rate, not activity.

Exposure Rate Calculations

The exposure rate at one meter above the ground plane is an accepted standard for fallout calculations. The exposure rate is caused by photons which directly impinge on the detector from the ground plane and also from radiation which has scattered before

reaching the detector. Three methods are available to calculate the contributions from the scattered radiation: Monte Carlo, successive scatters, and Build-Up Factors (BUF). This study examined successive scatters and BUF; a full Monte Carlo calculation was beyond the scope of this analysis.

The method of Build-Up Factors has been used by several others in previous work. Notably, Bigelow calculated BUF for a coarse mesh of energies ranging from 100 keV to 10 MeV (1:58-100), while Kalansky used a much finer mesh for energies below 1 MeV (11:140-141). Both authors used the method of moments expansion for their calculations. The primary limitation of the moments expansion method is the requirement for infinite homogeneous air, which does not accurately model the build up from a ground plane source of gamma-rays. Kalansky's values were used for the energies below 1 MeV while Bigelow values were used above 1 MeV. The BUFs were fit to three mean free paths using Taylor's BUF fit of the difference of two exponential functions:

$$A_1 \cdot e^{(C_1 X)} + A_2 \cdot e^{(C_2 X)} \qquad (4)$$

Note that the build-up factor must equal 1 at X=0, therefore $A_2 = 1 - A_1$. A least squares fit routine was used to determine the coefficients for the two exponential functions.

The BUF's treatment of the energy loss per collision and the resulting photon energy which ultimately strikes the detector can be

misleading. The energy build up is a measure of the extra energy deposited on a target in addition to the energy deposited by the photons that hit the target without scattering. The amount of extra energy deposited does not clearly establish the form of the energy deposited. For example, at a given number of mean free paths from a source of 1-MeV gamma rays, let us say that 1 out of 100 source photons directly hit the target and the energy BUF is 2. Therefore, in addition to the one direct photon that hits the target, an additional 1-MeV of energy is deposited by two or more photons (1 if Raleigh scatter is possible), which have scattered enroute to the target. The additional energy could come from two, 500-keV gamma rays, or three, 333-keV gamma rays, etc. The characteristics of the scattered photons are necessary to determine the absorption cross-section used at the detector. For this analysis the build up was simply used to increase the number of gamma rays of the original energy, i.e. no energy loss with collision. This assumption will underestimate the exposure at the target because the absorption cross section generally decreases with increasing energy up to a few MeV.

The method of successive scatters is well suited to problems in which the radiation rapidly attenuates. The model chosen for this study is an infinite homogeneous source of gamma rays, distributed over a smooth ground plane, where the ground is assumed to be an absolute absorber. Photon interactions can occur with air molecules above the ground, and the interaction rate is strictly a function of

altitude, i.e. the co-altitude reaction rate is constant. As a result, each altitude can be treated as a new infinite homogeneous plane of gamma rays, where the gamma-ray emission rate of this new source is the scattering rate at that altitude. The scattering source at altitude j is the number flux at altitude j times the scattering cross-section in air. The source can be written as:

$$S_j^1 = \sum_{i=1}^{19} NF_j^i \cdot (\mu_s^i)_{air} \qquad (5)$$

where

S_j^1 = Source created from the first scatter at altitude j

NF_j^i = Number flux from energy group i at altitude j

$(\mu_s^i)_{air}$ = Scatter cross-section for air at energy i

The number flux at altitude j is determined from the following equation:

$$NF_j^i = C \cdot GER_{total} f^i \int_0^{\infty} \frac{e^{-(\mu_t^i)_{air} s}}{4\pi s^2} 2\pi r\, dr \qquad (6a)$$

Which can be reduced to:

$$NF_j^i = \frac{C \cdot GER_{total} f^i}{2} \int_{(Z_j)}^{\infty} \frac{e^{-(\mu_t^i)_{air} s}}{s} ds \qquad (6b)$$

Where

GER = Gamma-ray emission rate per unit area in MCi

$C = 3.7 \cdot 10^{16}$ gamma ray emissions per MCi

f^i = Fraction of gamma rays in energy group i

$(\sigma_t^i)_{air}$ = Total cross-section for energy group i

Z_j = Altitude$_j$

r = Ground range

s = Slant range $= [(r^2)+(Z_j^2)]^{1/2}$

The integral is the exponential integral of the first kind, E_1, and the equation reduces to:

$$NF_j^i = \frac{C \cdot GER_{total}}{2} f^i E_1\left[(\mu_t^i)_{air} \cdot Z_j\right] \qquad (6c)$$

The scattered radiation sources are created at several altitudes for each energy.

The successive scatter method can be carried out to second, third, etc. scattering sources. The second-scatter source is created by the interactions of the first-scattered photons from all altitudes interacting with the air molecules at all other altitudes. This causes the calculation to increase by N^2, where N is the number of altitudes used. Likewise the third-scattered radiation is caused by interactions from the second-scattered radiation.

A complete Monte Carlo evaluation of the scattering process would track the photon's energy as it scattered, however the successive scatter calculation is not as flexible. Two necessary assumptions were made to produce a workable code: 1) the Compton scattered photons are isotropically scattered and 2) the energy lost during collision is always equal to the average energy loss as calculated from the Klein-Nishina approximation (9:687):

$$\frac{h\nu'_{ave}}{h\nu_0} = \frac{\sigma_s}{\sigma_t} \tag{7}$$

Where

$h\nu'_{ave}$ = Average energy of scattered photon

$h\nu_0$ = Initial photon energy

σ_s = Average scattering cross-section for energy $h\nu_0$

σ_t = Average total cross-section for energy $h\nu_0$

Appendix H lists the average energy of the scattered photon for the 18 energy bins used by DKPOWR. The isotropic scatter assumption probably increases the exposure rate slightly. The result of the average energy loss is unknown. The only way to fully resolve the error caused by these assumptions is by comparing results to those of a Monte Carlo calculation.

The exposure rate is calculated by summing interactions of direct and scattered photons in each energy group with the detector. The

exposure rate calculation is similar to determining the scattering source at various altitudes, however different conversions are used to calculate exposure rate rather than scattering rate. To calculate the exposure rate at the detector for the direct radiation, the following equations are used:

$$\dot{D} = C \cdot GER_{total} \sum_{i=1}^{19} f^i \int_0^{\infty} \frac{e^{-(\mu_t^i)_{air} s}}{4\pi s^2} \left(\frac{\mu_a^i}{\rho}\right)_{air} (h\nu^i) 2\pi r dr \quad (8a)$$

Which reduces to :

$$\dot{D} = \frac{C \cdot GER_{total}}{2} \sum_{i=1}^{19} f^i \left(\frac{\mu_a^i}{\rho}\right)_{air} (h\nu^i) \int_1^{\infty} \frac{e^{-(\mu_t^i)_{air} s}}{s} ds \quad (8b)$$

which further reduces to:

$$\dot{D} = \frac{C \cdot GER_{total}}{2} \sum_{i=1}^{19} f^i \left(\frac{\mu_a^i}{\rho}\right)_{air} (h\nu^i) E_1(\mu_t^i)_{air} \quad (8c)$$

Where

GER is in curies and

$$C = \left(3.7 \cdot 10^{10} \frac{\gamma ps}{Ci}\right)\left(3600 \frac{sec}{Hr}\right) \times$$

$$\left(1.602 \cdot 10^{-13} \frac{Joules}{MeV}\right)\left(\frac{1}{0.00877} \frac{Roentgens}{Joule/kg}\right)$$

$$= 2433 \frac{\gamma}{MeV} \cdot \frac{kg}{Ci} \cdot \frac{Roentgens}{Hr} \quad (9)$$

The contribution from the scattered photons is more complex to calculate. The method of successive scatters creates a series of planar sources of gamma rays at various altitudes, but in actuality a continuous vertical distribution is created by the scattering. The use of discrete altitudes is an approximation. At each discrete altitude sources of first, second, and third scattered photons exist. Therefore, instead of only using an area integral over a planar source of gamma rays, as the direct contribution is calculated, a volume integral is necessary to determine the contribution from the scattered photons. The discrete altitude approximation requires a numerical integration of the vertical component of the integral, while an integral over the source plane is calculated similar to the method used for the ground plane.

The exposure due to scattered radiation is calculated by:

$$\dot{D} = \sum_{k=1}^{3} \sum_{j=1}^{N} \left\{ \sum_{i=1}^{19} S_j^i \int_0^{\infty} \frac{e^{-(\mu_i' s)}}{4\pi s^2} \left(\frac{\mu_a'}{\rho} \right)_{air} (h\nu') r\,dr \right\} \Delta Z_j \qquad (10a)$$

By changing variables and rearranging, the equation is reduced to:

$$\dot{D} = \sum_{k=1}^{3} \sum_{j=1}^{N} \left\{ \sum_{i=1}^{19} \frac{S_j^i}{2} \left(\frac{\mu_a'}{\rho} \right)_{air} (h\nu') \int_{|Z_j - 1|}^{\infty} \frac{e^{-(\mu_i' s)}}{s} ds \right\} \Delta Z_j \qquad (10b)$$

Which can be expressed as:

$$\dot{D} = \sum_{k=1}^{3} \sum_{j=1}^{N} \{ \sum_{i=1}^{19} \frac{S_j^i}{2} \left(\frac{\mu_a^i}{\rho} \right)_{air} (h\nu^i) \, E_1(|Z_j - 1| \cdot \mu_t^i) \} \Delta Z_j \qquad (10c)$$

where the outside summation accounts for first, second and third scattering, the next summation is the vertical numerical integration, and the inner summation accounts for the 19 energy groups. Z_j is altitude j and ΔZ_j is the vertical width associated with altitude j. All other terms were described for the ground calculation.

The Source Conversion Coefficient (SCC) is the exposure rate (Roentgens/Hour) at a detector 1 meter above the ground, due to a source emitting 3.7×10^{18} gamma-rays per second spread evenly over a smooth surface 1 km^2. The SCC is a useful parameter because the only fission product characteristic used in its calculation, is the gamma-ray spectrum as a function of time. The gamma-ray emission rate is a constant, therefore the time dependence of the fission product decay is removed.

The Source Normalization Constant commonly referred to as "K" is routinely used in fallout and exposure rate calculations. Common units of K are (Roentgens/Hour)/(kT/km^2) and, by convention, K is determined for the gamma-ray emission rate and gamma-ray energies present at one hour following the weapon detonation. K is determined

by multiplying the SCC(1Hr)*GER(1Hr). In addition, a time dependent Source Normalization "Constant" can by calculated by multiplying the SCC at time t times the GER remaining at time t.

III Methods of Analysis

The first step in the analysis was to run DKPOWR and ORIGIN2 for the appropriate fuel and display times. DKPOWR calculated all its data on a single fission, while ORIGIN2 determined the data as a function of reactor power output. The amount of prompt and delayed energy released by a fission event was needed to convert both sets of data to a "per kiloton of fission yield" basis. The ORIGIN2 data was simply multiplied by the ratio of prompt energy/(prompt + delayed energy). This ratio is used because the power reactor definition of yield is the combination of both the prompt heat and radiation along with the delayed decay energy, while the weapon definition of yield is the prompt energy alone. In a personal conversation with George H. Nickel, Donald Barr of the INC Division of Los Alamos National Laboratory, stated that the prompt energy released from a single U-235 fission is taken as 183 MeV (12:*) while the prompt release from Pu-239 is 185 MeV (2:2.3) and 183 MeV from U-238. Total yields of 200 MeV for both uranium isotopes and 205 MeV for plutonium were used (2:2.3). The resulting conversion factors for the number of fissions per kiloton are $1.428X10^{23}$ for U-235 and U-238, and $1.412X10^{23}$ for Pu-239.

For calculations with ORIGIN2, a fast breeder reactor configuration was used with a photon library that did not include water or bremsstrahlung. These interactions were omitted since water is not

present in a nuclear weapon and the air surrounding the weapon is sufficiently sparse that bremsstrahlung should not be significant. DKPOWR used thermal fission for U-235 and Pu-239 and fast fission for U-238. While minor differences are present due to various neutron energies, the differences between fuels is more important (7:*). The differences due to neutron energy occur primarily in the lower fission yields and probably does not effect the overall decay characteristics of the fission products. An analysis which includes fast neutron cross-section data should clarify any discrepancies which may occur.

Both DKPOWR and ORIGIN2 were created for reactor work and a weapon had to be modeled within the working parameters of each code. For example, two input parameters needed for DKPOWR were pulse length and number of fissions per second. A pulse length of 10^{-04} seconds along with 100^{4} fissions per second resulted in one fission taking place during the pulse. The output was easily scaled to appropriate values. Two similar parameters were required for ORIGIN2, the pulse length and reactor power level. The reactor was pulsed at a power level of 500,000 MW for 20 seconds, resulting in a prompt yield of approximately 2.1 kT.

DKPOWR presented the activity and the total number of MeV emitted in each energy bin as a function of time. The total number of gamma rays emitted per second was calculated by dividing the MeV/s from each bin by the bin's midpoint energy and summing over all the

bins. By taking the ratio between the activity and the total number of gamma rays emitted per second (GER), the number of gamma rays emitted per disintegration can be determined. Although the activity of the fission products is more commonly referred to, the total number of gamma rays emitted per second is of more interest in fallout studies. Due to the short mean free path of the β^- particle, contamination from the gamma-ray emitting isotopes is of primary concern in fallout calculations.

The energy spectrum from DKPOWR is presented in the output with the endpoints energies of the energy groups, while ORIGIN2 lists the midpoint energy. For this analysis, the output of an energy group was assumed to be associated with the midpoint energies. This assumption probably underestimates the absorption cross section in the first energy bin, Ø-1ØØ keV. As a result, the attenuation for that energy group is underestimated and its contribution in the conversion to exposure rate will be overestimated. The relative change of the cross-sections over the width of the other energy groups should not produce a significant error. The absorption and total cross-sections for air were interpolated from a graph in Rockwell's **Reactor Shielding Manual** (13:448-45Ø). The scattering cross-section for air was determined simply as the difference between the total and absorption cross-sections. Pair-production cross-sections for air were interpolated from Evans' **The Atomic Nucleus** (9:689). The absorption and scattering cross-sections were determined more accurately

than the pair-production, because of the readability of the respective graphs. However, since very few photons of sufficient energy are emitted, the effect of pair production is minimal, and therefore is not included in this study.

The decay of the activity and GER with time was analyzed by running the DKPOWR code for several different times and analyzing the calculated results. The one hour GER value was used along with the Way-Wigner $t^{-1.2}$ relation to calculate the approximate GER for times both prior to and after one hour. This assured that both the DKPOWR calculated GER and the approximated GER were based on the same reference and a direct comparison could be made.

The energy spectrum is used as input for the successive scatter and Build Up Factor (BUF) calculations. The bulk of the calculations in this study were based on the method of successive scatter. The first, second, and third scatter sources were calculated at altitudes up to 150 meters. The first scatter contributed average of about 4.8% of the direct, second scatter 0.9%, and third 0.3%. The calculation was performed at altitudes up to 1000 meters. Limiting the calculations to 150 meters introduced less than 0.1% error and significantly reduced the computer run time.

The exposure rate contributions from the direct and each of the scatter sources were calculated by using the Source Conversion Coefficient (SCC) because it is independent of the (GER) from the fission

products released by the weapon and allows the GER calculations to be analyzed separately. The SCC could be incorporated into other calculations in which the user would apply his own GER.

The Source Normalization Constant (K) is a conversion factor from GER per unit area to exposure rate. By convention, K is determined as if the total GER released from the weapon were spread homogeneously over a 1-km^2 area, one hour after the detonation. Time dependence is simply governed by a separable $t^{-1.2}$ factor. The Source Normalization Constant was determined by using the SCC calculated from the above methods along with the GER calculated by DKPOWR. A time dependent Source Normalization "Constant" is easily determined by the following relation:

$$K(t) = \mathrm{SCC}(t) \cdot \mathrm{GER}_t(t) \qquad (11)$$

where

GER$_t$ = Total GER released by the weapon at time t in MCi

The exposure rate calculations will be adjusted by the local fallout fraction, ground roughness factors, wind patterns, etc.

IV Results and Discussion

Fission Product Decay

The total activity and the gamma-ray emission rate (GER) from the fission products released by the weapon varies weakly with the fuel type. Table 1 shows the GER determined from DKPOWR, ORIGIN2, and Glasstone's reference at one hour after detonation. The number of gamma rays emitted per disintegration calculated from DKPOWR results was used to convert both the DKPOWR and ORIGIN2 results to GER. As can be seen from Table 2, both codes produced activity results which compare very well. The difference between the values for U-235 activity is less than 3%. Glasstone makes no distinction between fuels, but states that approximately 530 Gamma-Megacuries of activity is released per kiloton of fission yield. As noted earlier, the term Gamma-Megacuries is taken as meaning 3.7×10^{16} gamma-ray emissions per second. The relationship between activity and GER is the average number of gamma rays emitted per disintegration. DKPOWR results were used to compare the total number of gamma rays emitted to the activity or total number of disintegrations. As can be seen from Figures 1, 2, and 3 the ratio changes dramatically as a function of time but the changes are similar for each fuel.

Table 1
GER Per Kiloton One Hour After Detonation, Gamma Mega Curies

FUEL	DKPOWR	ORIGIN2	GLASSTONE
U-235	591	607	530
Pu-239	590	605	530
U-238	643	652	530

Table 2
Activity Per Kiloton One Hour After Detonation, MCi

FUEL	DKPOWR	ORIGIN2	GLASSTONE
U-235	435	447	*****
Pu-239	402	412	*****
U-238	434	440	*****

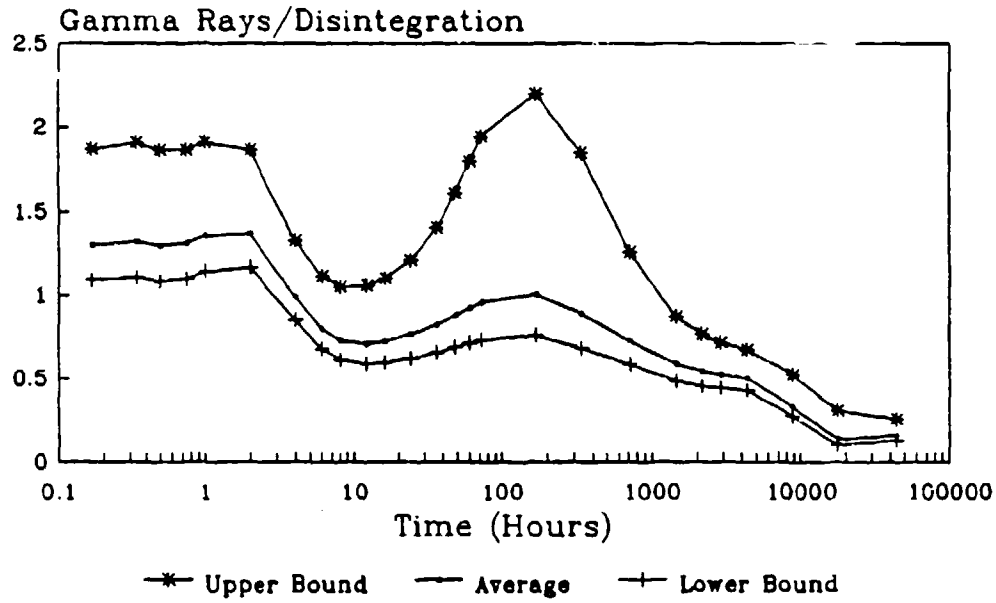


Figure 1
Gamma Rays per Disintegration as a Function of Time for U-235

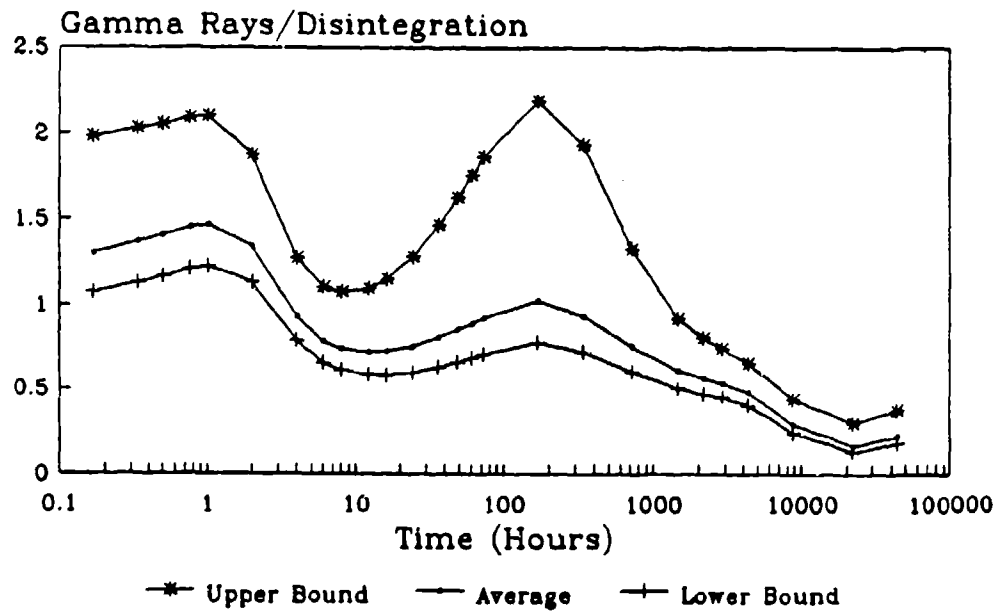


Figure 2
Gamma Rays per Disintegration as a Function of Time for Pu-239

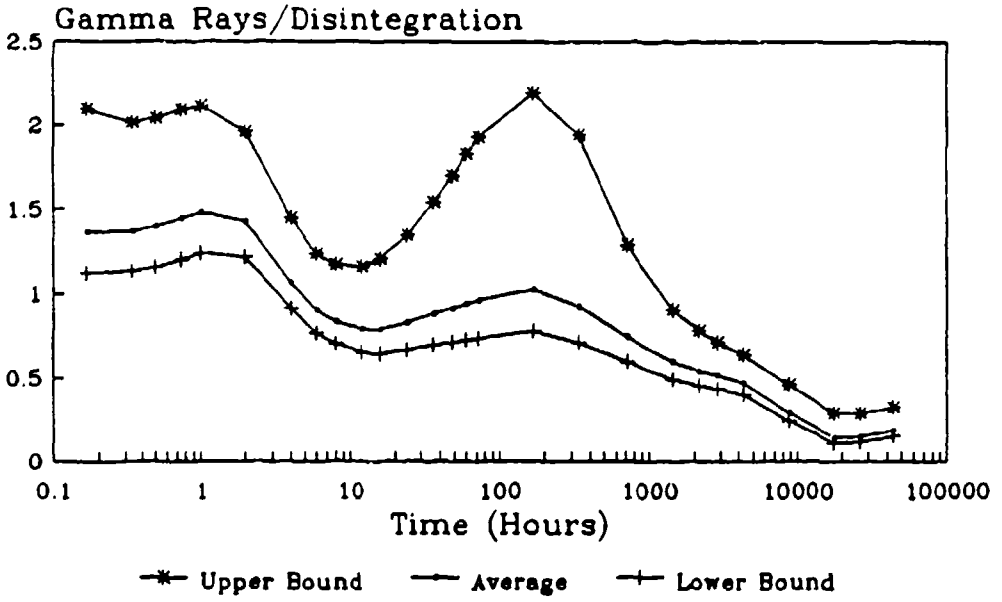


Figure 3
Gamma Rays per Disintegration as a Function of Time for U-238

As can be seen from Figures 1, 2, and 3, substantial error margins are associated with the calculation of the number of gamma rays per disintegration. The spectral data from DKPOWR is in MeV per second in each energy bin. To determine the number of gamma rays emitted per second, the MeV per second value was multiplied by the average energy of the bin. As noted earlier, the average energy was taken as the midpoint of the bin. In actuality the average energy of each bin may be something other than the midpoint, however, the precise value is impossible to determine from the DKPOWR output. To determine an error bar for the gamma-ray per disintegration value, the number of gamma rays was determined by dividing by the minimum value of the energy bin, and by the maximum value. For the first bin the minimum is Ø, therefore an arbitrarily value of Ø.1 was chosen for

the minimum. Obviously the uncertainty in the minimum of the first energy group will have a dramatic effect on the upper bound of the error. The upper bound value ranged from 40-120% above the midpoint value and lower bound was consistently about 16-25% below the mid-point value.

Gamma-ray spectra were calculated for the U-235, U-238, and Pu-239 fuels at times ranging from 10 minutes to 5 years using DKPOWR. The spectra were calculated for 100% pure fuel. The spectra from the U-235 fuel is presented in Appendix A, Appendix B shows the Pu-239 data and Appendix C shows the U-238 data. The spectra have been normalized to a total of 1 gamma-ray per second per MeV.

As noted earlier, the U-235 and Pu-239 data have been supplemented with experimental results at early times. As can be seen in the early spectra, the high energy photons present in the U-235 and Pu-239 fission products are missing from the U-238 data. As a consequence, the early time average energy for U-238 is significantly lower than both the other fuels, which can clearly be seen in Figure 4 which presents the average energy for all three fuels from 10 minutes to 5 years. For times greater than about 1 hour all three fuels show the same general trends in the shape of the average energy vs time curve.

The average energy, at 1000 seconds for U-235 fuel was calculated by R. B. Drinkwater as 1054 keV (5:30). Drinkwater also noted that the Air Force Weapons Laboratory's Firefly study determined an

energy of 848 keV (5:30). Table 3 presents the values calculated from this study, along with values from Drinkwater and Firefly. The average was determined by summing the total number of MeV of energy emitted per second and dividing by the total number of gamma rays emitted per second. The DKPOWR average energy for U-235 compares within 0.5% of the value from the Firefly study. Glasstone suggested average energy of 700 keV at one hour is compared to DKPOWR and ORIGIN2 values in Table 4. The values from DKPOWR and ORIGIN2 compare within 2% for U-235 and less than 0.25% for Pu-239 at one hour. However at 1000 seconds a discrepancy of approximately 10% exists for both U-235 and Pu-239. As noted earlier DKPOWR is more reliable at shorter times, i.e. less than about 20 minutes, due to the inclusion of experimental data supplements for these two fuels and due to the 20 second pulse used by ORIGIN2. The difference between the two codes for U-238 calculations is not as great, (approximately 2% at 1000 seconds and 0.5% at 1 hour), because no experimental data was used for DKPOWR. Since ORIGIN2 uses fast reactor neutron energies and DKPOWR uses thermal neutrons, it should be noted that the agreement between the codes for U-235 and Pu-239 at one hour supports the assumption that the neutron energy difference produces a negligible error.

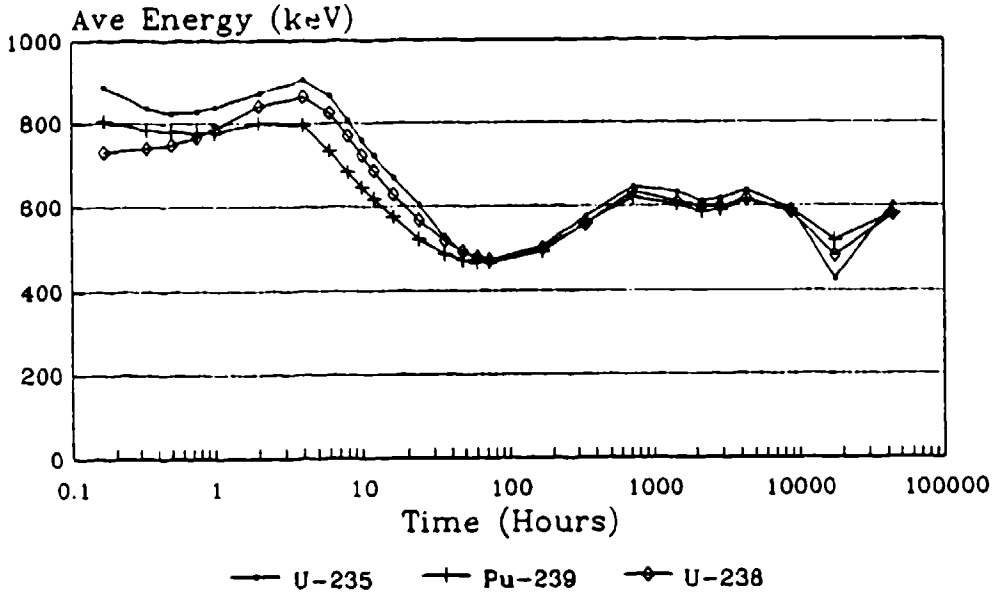


Figure 4
Average Energy as a Function of Time

Table 3
Average Energy at 1000 Seconds (keV)

FUEL	DKPOWR	ORIGIN2	DRINKWATER	FIREFLY
U-235	852	761	1054	848
Pu-239	790	713	****	***
U-238	738	723	****	***

The Way-Wigner approximation for total fission product decay follows the relationship of $t^{-1.2}$. Glasstone stated that the relationship was invalid for times shorter than 30 minutes and longer than 6

Table 4
Average Energy at One Hour (keV)

FUEL	DKPOWR	ORIGIN2	GLASSTONE
U-235	839	823	700
Pu-239	777	776	700
U-238	785	781	700

months. The calculations Glasstone references include over 300 isotopes of fission products and induced activity along with fractionation and partial loss of gaseous products (10:450). The differences between Glasstone's reference and this study preclude any exact comparisons, however, the data from fission products only show the Way-Wigner approximation to be accurate for activity decay within approximately 25%, except for a period from 1 to 6 months where the difference reaches 45%, to at least to 5 years. However, the curve associated with the decay of the number of gamma rays emitted per second does show a divergence similar to Glasstone (10:392-393). Figure 5 displays the decay of the activity as a function of time for U-235 fuel, along with Way-Wigner's approximation. The discrepancy from 1 to 6 months reaches a largest difference of 45% at 4 months. Similar results for Pu-239 and U-238 data are shown in Figures 6 and 7 respectively. Figures 8, 9 and 10 present the results of the decay of the GER as a function of time as compared to Way-Wigner's approximation. The divergence at approximately 6 months is caused by

decrease in the number of gamma rays emitted per disintegration. At 5 years only one gamma ray is emitted for approximately every 8 decays.

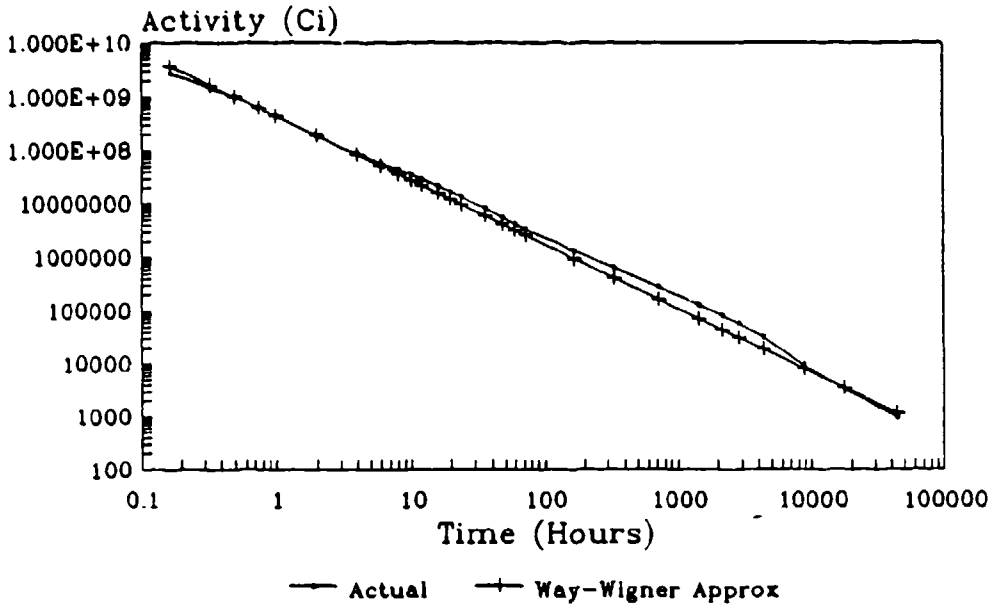


Figure 5
Calculated Activity vs the Way-Wigner Approximation for U-235

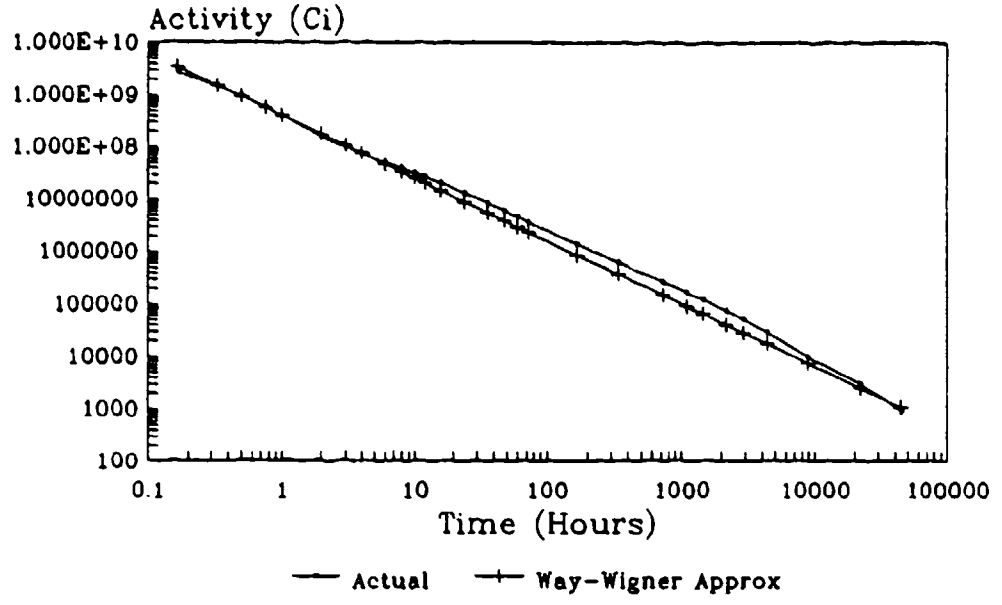


Figure 6
Calculated Activity vs the Way-Wigner Approximation for Pu-239

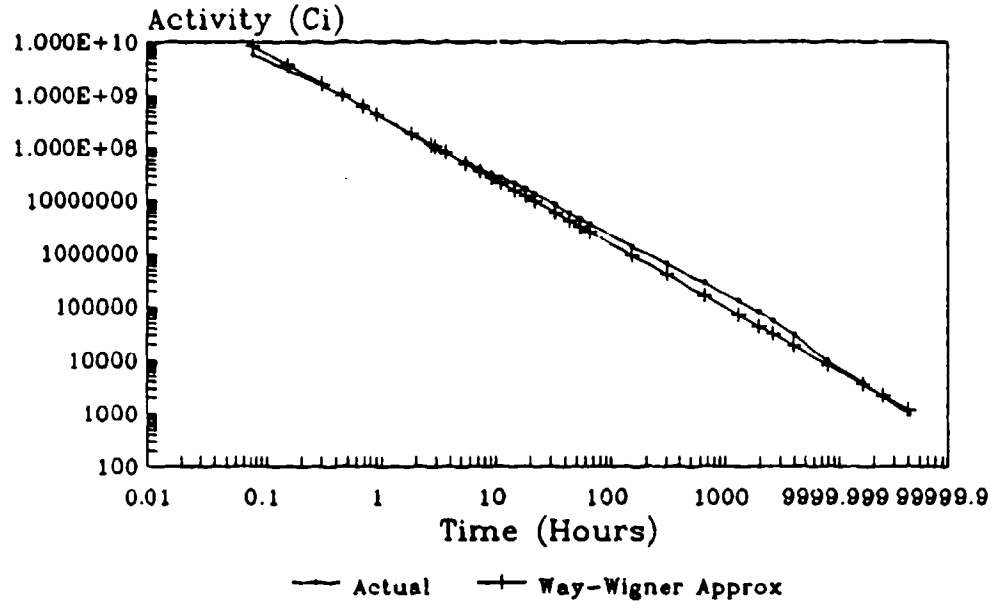


Figure 7
Calculated Activity vs the Way-Wigner Approximation for U-238

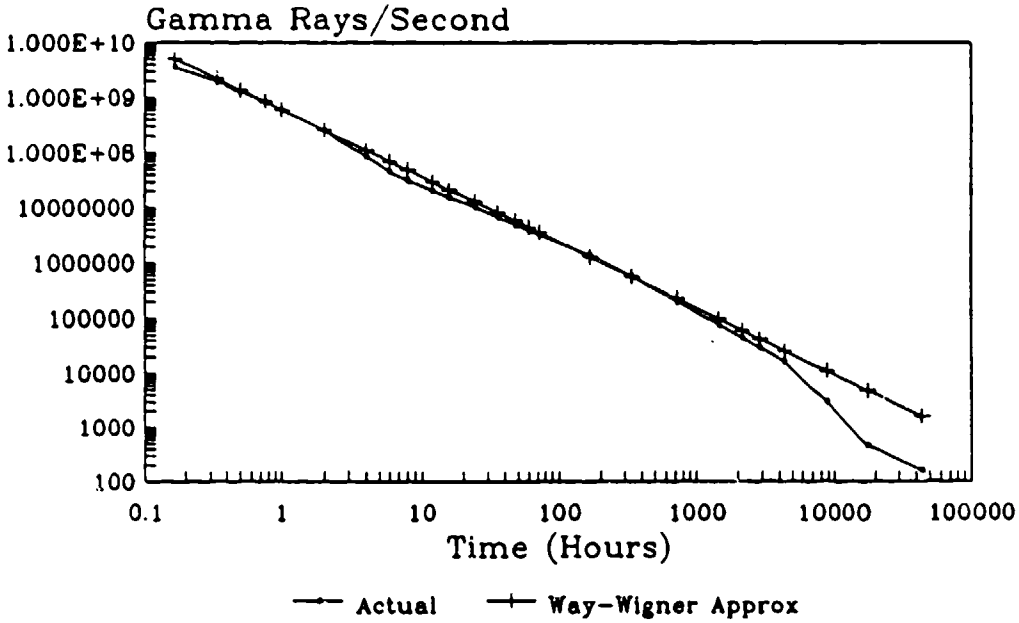


Figure 8
GER vs the Way-Wigner Approximation for U-235

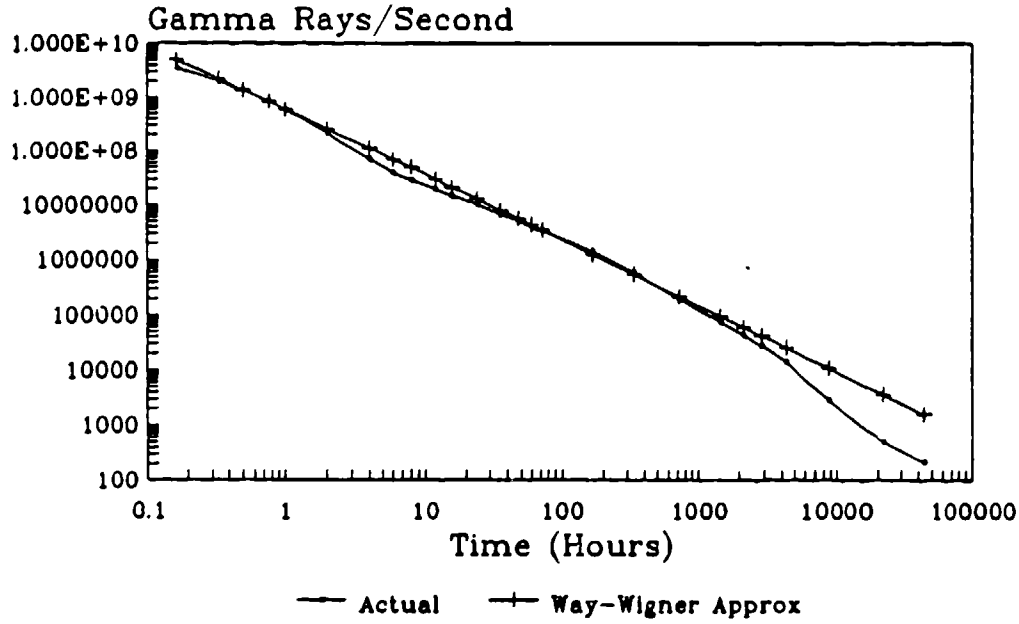


Figure 9
GER vs the Way-Wigner Approximation for Pu-239

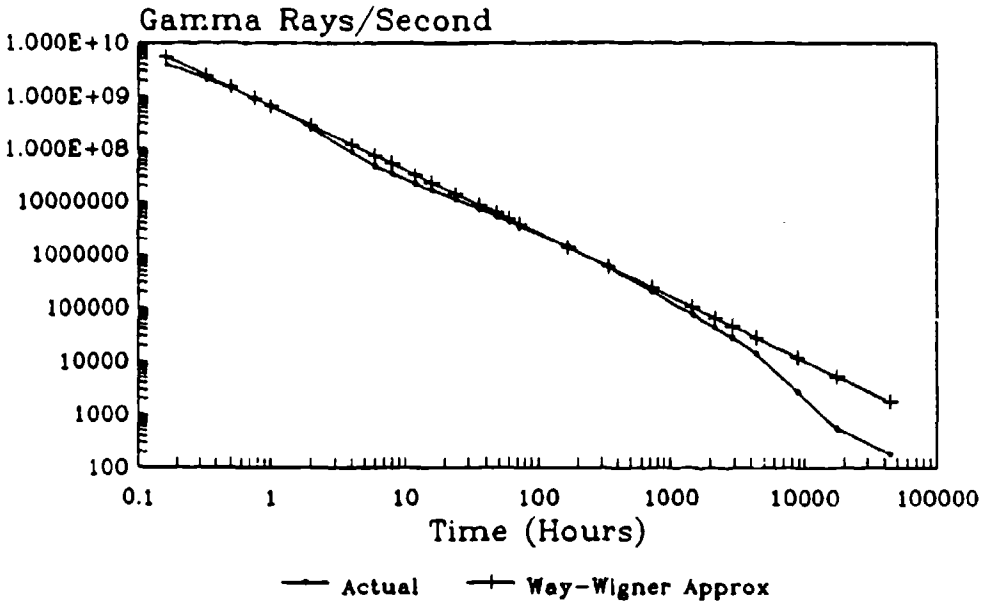


Figure 10
GER vs the Way-Wigner Approximation for U-238

Conversion to Exposure Rate

The Build Up Factor (BUF) and successive scatter calculations were performed on a desk top computer, using data from DKPOWR output. The BUFs were determined from Bigelow and Kalansky's earlier works, and were fit to Taylor's BUF function, Equation 3. The coefficients C_1, C_2, A_1 and A_2 are listed in Appendix F.

The results of the BUF and successive scatter calculations differed by a substantial margin, the BUF was consistently approximately 14% higher. One reason is that the moments expansion necessarily used infinite homogeneous air to determine the build up factors. The successive scatter method assumed that the media below the ground plane was an absolute absorber. As a result, contributions from

"imaginary" air below the ground plane in the BUF method was contributing to the scattered radiation. To check the contribution due to the "imaginary" air, the exposure rate was calculated using the successive scatter method, with no energy loss with collisions, with air both above and below the ground plane. The method of BUF still resulted in an exposure rate that was 9% higher than the successive scatter calculation. The method of successive scatters was believed to be based on more valid approximations than the BUF method, and was used for the bulk of the data.

The validity of the assumption that pair-production made a negligible contribution was also performed. The 10-minute U-235 spectrum, which contained the largest number of high energy photons, and the 1-hour U-235 spectrum were used to determine the Source Conversion Coefficient with the pair-production cross-section providing an additional source term of two photons in the 500-600 keV energy group per pair production interaction. The result was a decrease in the SCC of approximately 1.3% at 10 minutes and less than 1% at 1 hour. Based on these minor differences and the poor accuracy of the pair-production cross sections, pair-production was ignored in the rest of this study.

Both the BUF and successive scatter methods were used to calculate the Source Conversion Coefficient (SCC) for monoenergetic gamma rays, to compare with a graph shown in Glasstone's **Effects of Nuclear Weapons**. Glasstone presents the exposure rate at various altitudes

above the ground plane and states that the data includes build-up due to scattered radiation (10:453-454). Figure 11 shows Glasstone's exposure rate at 1 meter above the ground, along with the monoenergetic calculations from successive scatter and BUF. The graph clearly shows the similarity between Glasstone's values and the values obtained from infinite homogeneous Build-Up Factors. Glasstone's lack of references prohibit verification of the methods used for his data.

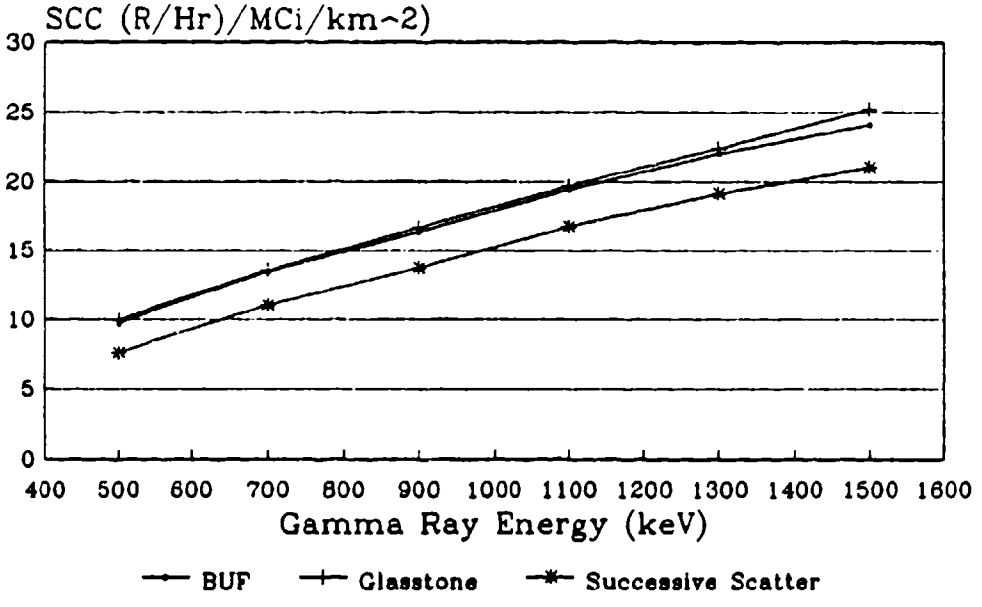


Figure 11
Mono-Energy Gamma Ray Source Conversion Comparison

The Source Conversion Coefficient is used because it is independent of the changes of the GER with time. The SCC is based on a source of 3.7×10^{16} gamma rays per second homogeneously spread over 1 km^2. As a result, the SCC is a convenient method to examine the exposure rate changes due to energy spectrum fluctuations. Figure 12

displays the Source Conversion Coefficient as a function of time, and if this graph is compared to Figure 4, the time dependence of the average energy, it can be noted that both curves exhibit similar trends.

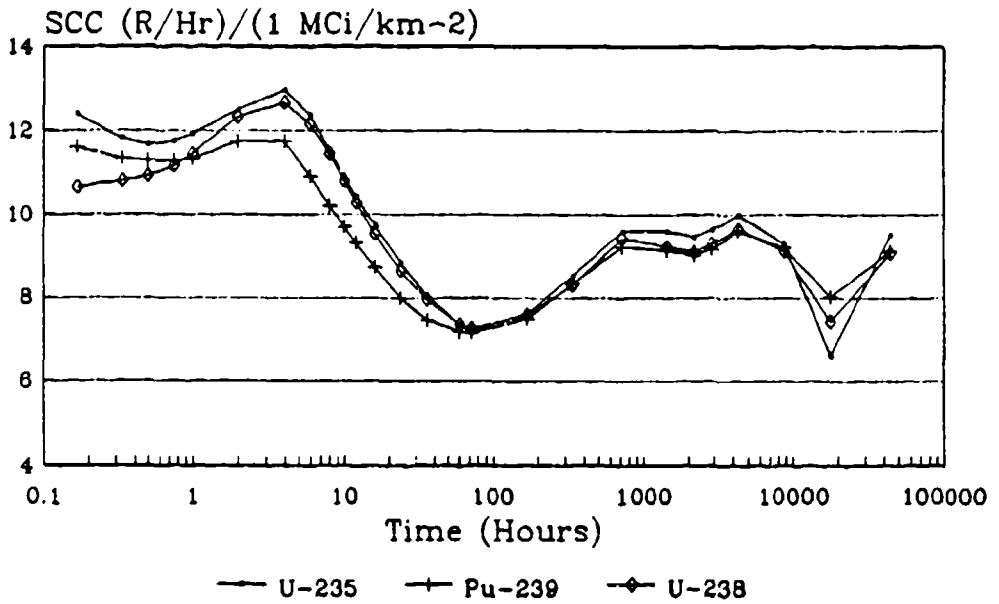


Figure 12
Source Conversion Coefficient as a Function of Time

The Source Normalization Constant (K) is the most commonly used conversion between GER and exposure rate. The conversion may include corrections such as the local fallout fraction and the ground roughness factor. The historic interpretation of K is based on the GER and average gamma-ray energy at one hour after the fission event and the time dependence of the GER is separated using the Way-Wigner approximation. In Table 5 the Source Normalization Constant at one hour is presented along with Glasstone's value. A time dependent Source Normalization "Constant" can be calculated by

multiplying the time dependent SCC and GER. The time varying Source Normalization Constants presented in Figures 13, 14, and 15 are based on the GER and energy spectrum present at the corresponding time. Figure 13 presents the Source Normalization Constant as a function of time for U-235 fuel. Pu-239 and U-238 results are presented in Figures 14 and 15 respectively. As expected, K follows the same general trends as the GER decay. Along with the time dependent K, the value at 1 hour with Way-Wigner decay is shown in these figures. Only first order approximations to exposure and exposure rate should be calculated using K and the Way-Wigner approximation. Even prior to 6 months the actual and approximated values differ by up to 85%. The Way-Wigner approximation should not be used for anything but a gross estimate of exposure rate for times greater than 6 months. A more accurate calculation of the exposure, at times both prior to and after 6 months, can be evaluated by numerically integrating the time dependent GER and SCC.

Table 5
Source Normalization Constant at One Hour After Detonation

Fuel	R/Hr/(kT/km^2)		R/Hr/(kT/Mi^2)	
	DKPOWR	Glasstone	DKPOWR	Glasstone
U-235	7Ø59	7186	2749	28ØØ
Pu-239	6684	7186	26Ø4	28ØØ
U-238	7369	7186	2871	28ØØ

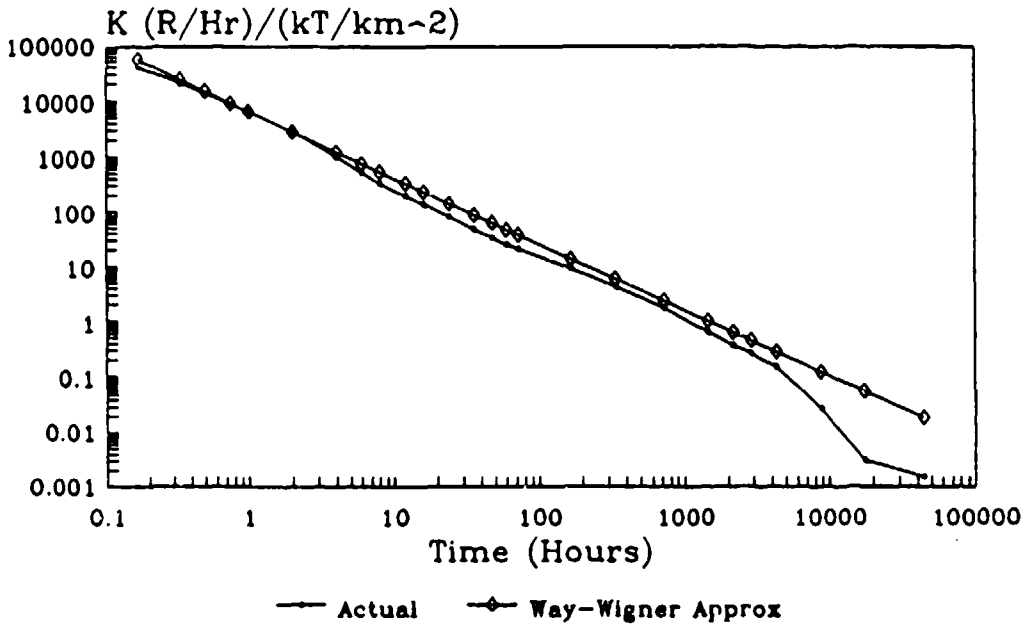


Figure 13
U-235 Source Normalization Constant vs Way-Wigner Approximation

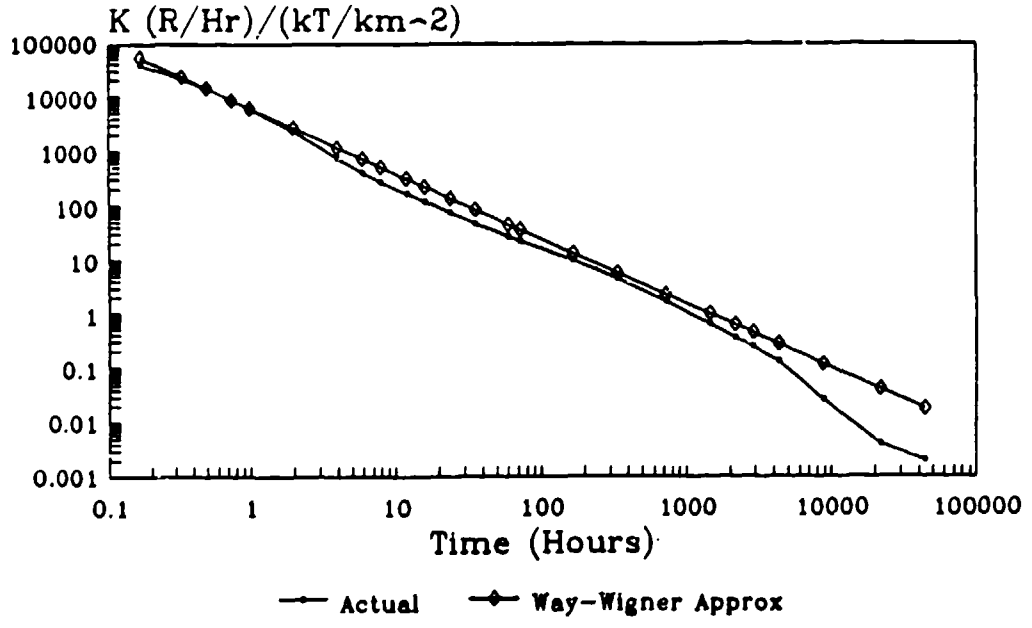


Figure 14
Pu-239 Source Normalization Constant vs Way-Wigner Approximation

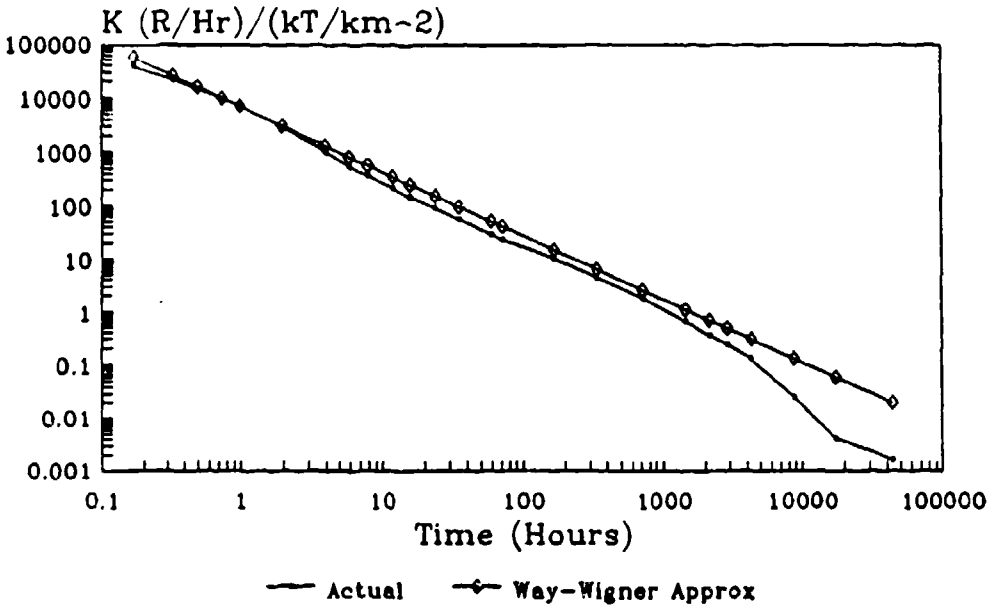


Figure 15
U-238 Source Normalization Constant vs Way-Wigner Approximation

V Conclusions and Recommendations

The purposes of this study were:

(1) Determine total activity and gamma ray emission rate of the fission products following a nuclear detonation.

(2) Examine the energy spectrum of the photons emitted from the fission products as a function of time.

(3) Compare the time varying results with Way-Wigner's approximation for fission product decay.

(4) Determine the effects the energy spectrum and total gamma ray emission rate have on the exposure rate calculations performed in fallout codes.

The activity, gamma-ray emission rate (GER) and energy spectra were determined using DKPOWR, a non-linear least squares approximation to output from experimental data and CINDER-10, an isotope generation and depletion code used by Los Alamos National Laboratory. Additional comparisons were made to results from ORIGIN2, an isotope generation and depletion code created at Oak Ridge National Laboratory for reactor studies. The energy build up due to photon scatters in the air were determined using the method of successive scatters.

The fission product activity and gamma-ray emission rate are not equivalent. Isotopes may emit none, one or several gamma rays each time they disintegrate. Since gamma rays are of primary concern in

fallout calculations, the GER should be used instead of activity. The ratio of GER/(disintegrations/second) was used to determine the average number of gamma rays emitted from a disintegration. A large upper and lower bound to the error associated with the calculation of gamma rays/disintegration exists because the spectral data from DKPOWR is presented as MeV/second in a given energy bin. To determine the number of gamma rays emitted, the number of MeV/second is divided by the average energy of the bin. The midpoint of the bin was used in this study, however an error bar is associated with the calculation because the average may be something other than the midpoint. To bound the error, the number of gamma rays was calculated by dividing by the minimum energy of each bin (upper bound) and the maximum energy (lower bound). The minimum energy in the first bin is Ø, therefore an arbitrarily chosen value of Ø.Ø1 Mev was used. As a result, the upper bound for the error is much larger than the lower bound.

The energy spectrum also changes with time. Initially, the spectrum is broad, ranging from a few keV to several MeV, however by 1 hour after the detonation the highest significant contribution is less than 3 MeV. The energy spectrum is fuel dependent, however some basic trends are evident in all three fuels, U-235, Pu-239, and U-238.

The rapid drop in the accuracy of Way-Wigner's approximation at 6 months is shown in the results of this study for the GER as a function of time. The activity of the fission products does not exhibit

the dramatic change, but remains accurate to within about 45% for times up to 5 years. As a result, as Glasstone suggested, the Way-Wigner approximation for GER is not valid for times greater than 6 months, however it is a good first order approximation for the activity decay. The activity and GER do not decay at the same rate because the number of gamma rays emitted for each disintegration is not a constant. The ratio drops significantly at times greater than about 6 months. At 5 years after the burst, 1 gamma ray is emitted for about every 8 disintegrations.

The effect of the energy spectrum variations is readily apparent in the Source Conversion Coefficient which is determined assuming a source of 3.7×10^{18} gamma rays per second is distributed over 1 km^2. The average gamma-ray energy and the SCC curves follow the same basic trends. The Source Normalization Constant (K) is dependent on the total number of gamma rays emitted by the fission products and the time behavior of their decay. The Source Normalization Constant at one hour determined by this study, is 7059 R/Hr/(kT/kM^2) for U-235, 6684 for Pu-239 and 7369 for U-238 (+45%/-16%). These results are within 7% of Glasstone's value of 7186. The data from this study do not include the effects of induced activity or the loss of activity due to gaseous products.

The historic method of determining exposure rate calculations is by using the Source Normalization Coefficient, which is valid at one hour, and applying Way-Wigner's $t^{-1.2}$ approximation for fission product

decay to convert to other times of interest. This method is valid only as first order approximation to calculate the exposure rate to about 6 months, because differences up to 85% can be seen between this method and calculating the time dependent Source Normalization "Constant". At 6 months, the GER diverges significantly from the $t^{-1.2}$ approximation, and the Way-Wigner approximation applied to K should only be used as a gross estimate. Accurate exposure calculations should be accomplished by numerically evaluating the time dependent GER and SCC, not by using the Way-Wigner approximation.

Several areas of this study should be examined further. A full Monte Carlo treatment of the gamma-ray scattering would provide needed insight into the accuracy of the successive scatter method used in this study. The assumptions of isotropic scatter and average energy loss for every collision have introduced unknown errors into the nature of the contributions from the scattered photons. In addition, the effects of the air-ground interface and ground scattering should also be examined.

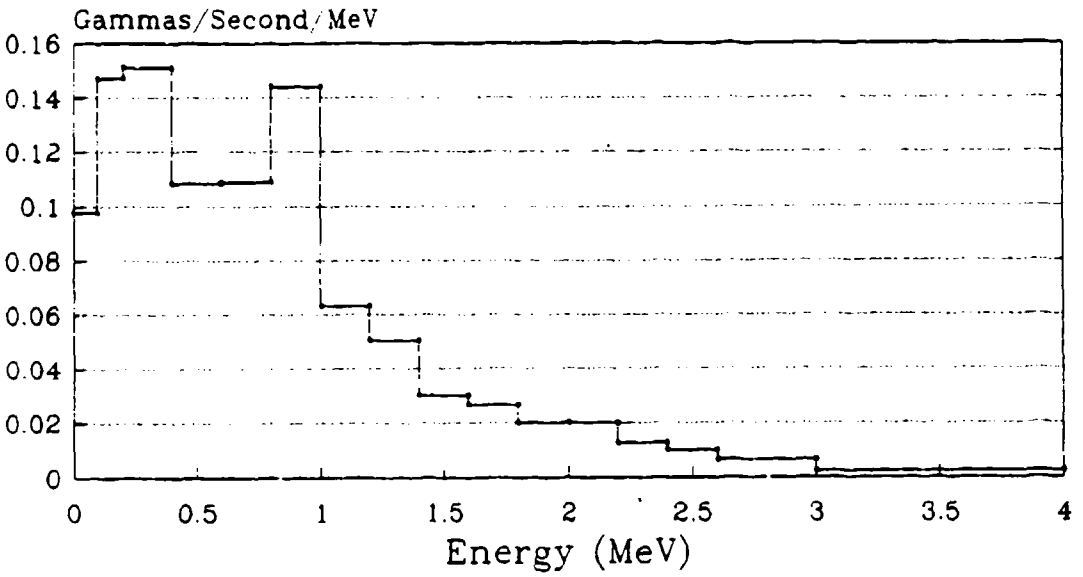
An examination of the isotopes remaining at various times would provide insight into the effects of factors such as the loss of gaseous fission products which should not be included in ground contamination. In addition, the effects of induced activity, both fission product, weapon case, and environmental materials should be examined.

Appendix A

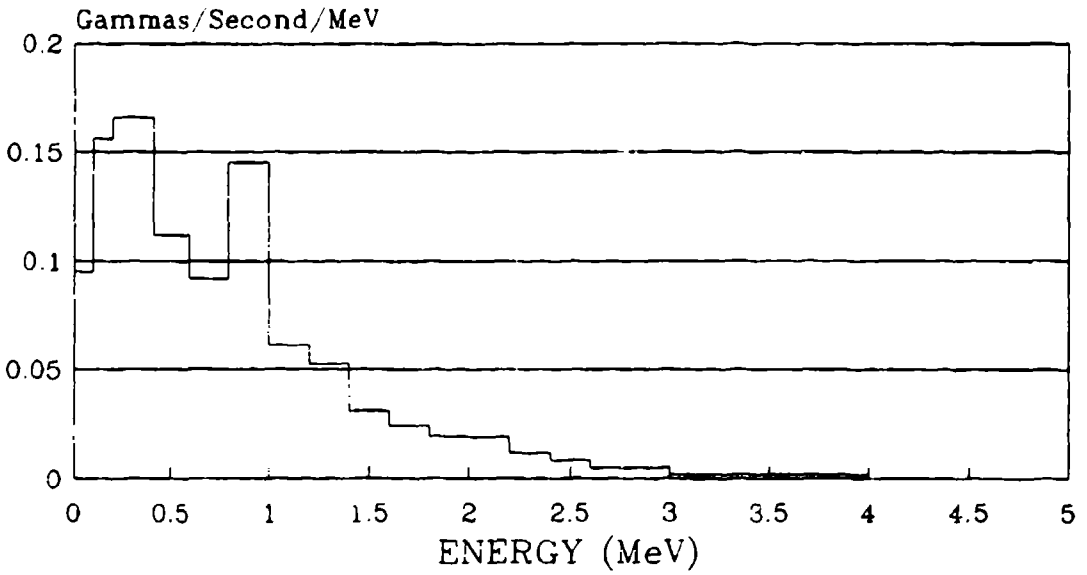
U-235 Fission Product Gamma Ray Spectra From 10 Minutes to 5 Years

The following spectra are from the fission products from U-235 fuel. The spectra have been normalized to one gamma-ray per second per MeV.

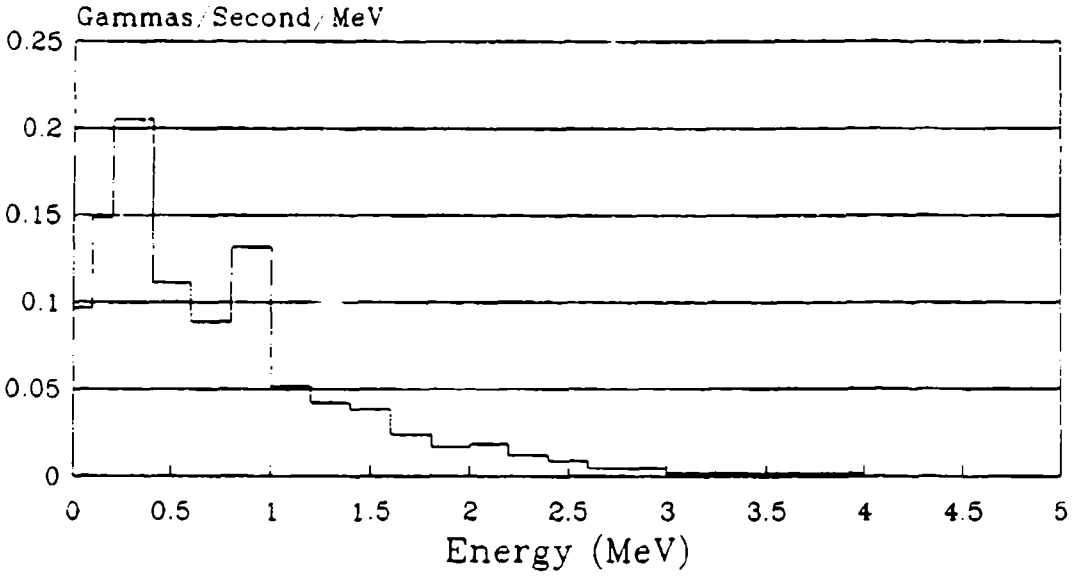
Time = 10 Minutes
U-235



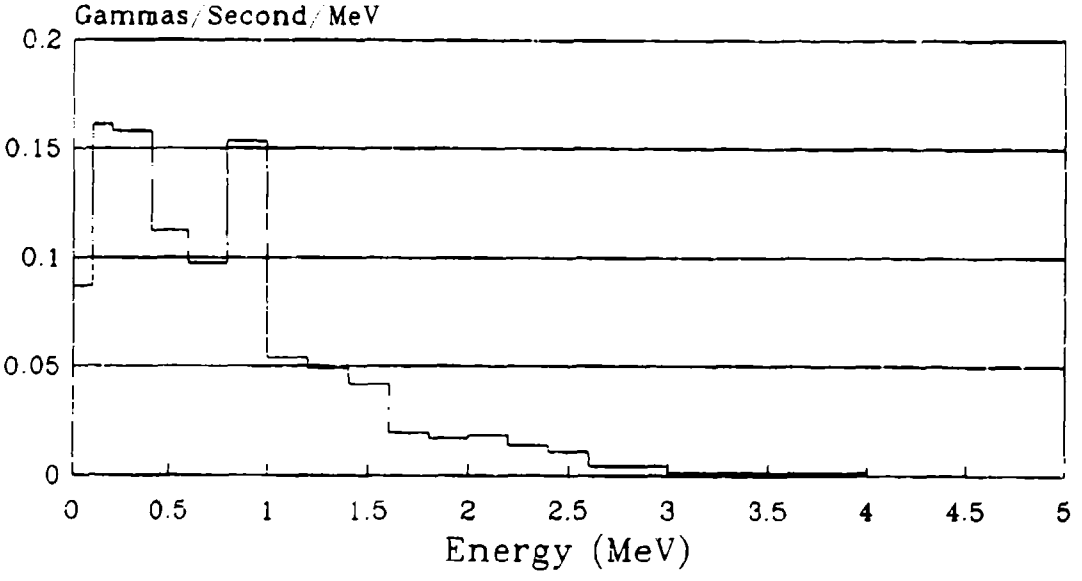
Time = 20 Minutes
U-235



Time = 30 Minutes
U-235

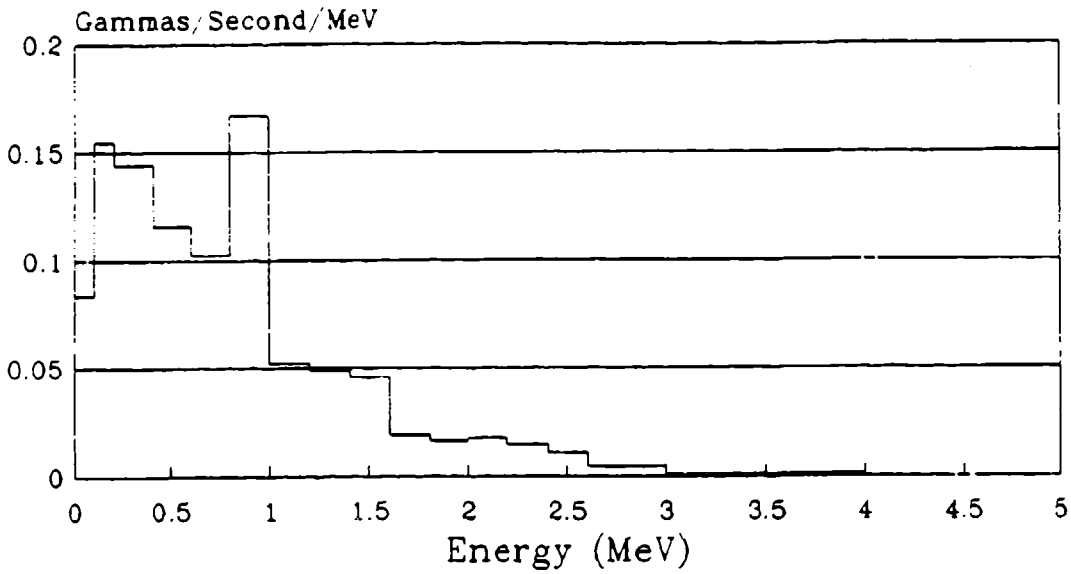


Time = 45 Minutes
U-235



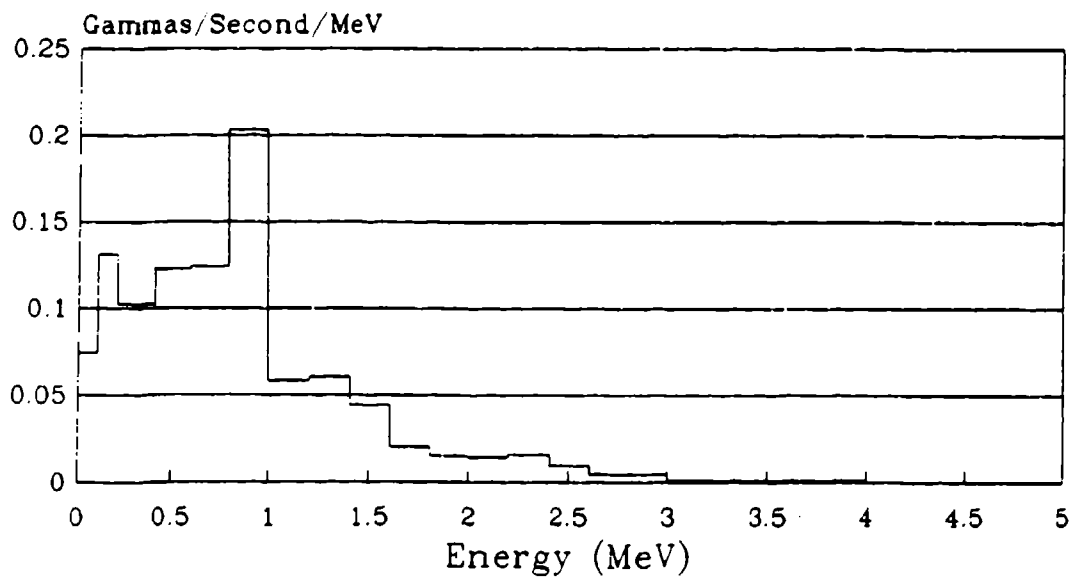
Time = 1 Hour

U-235

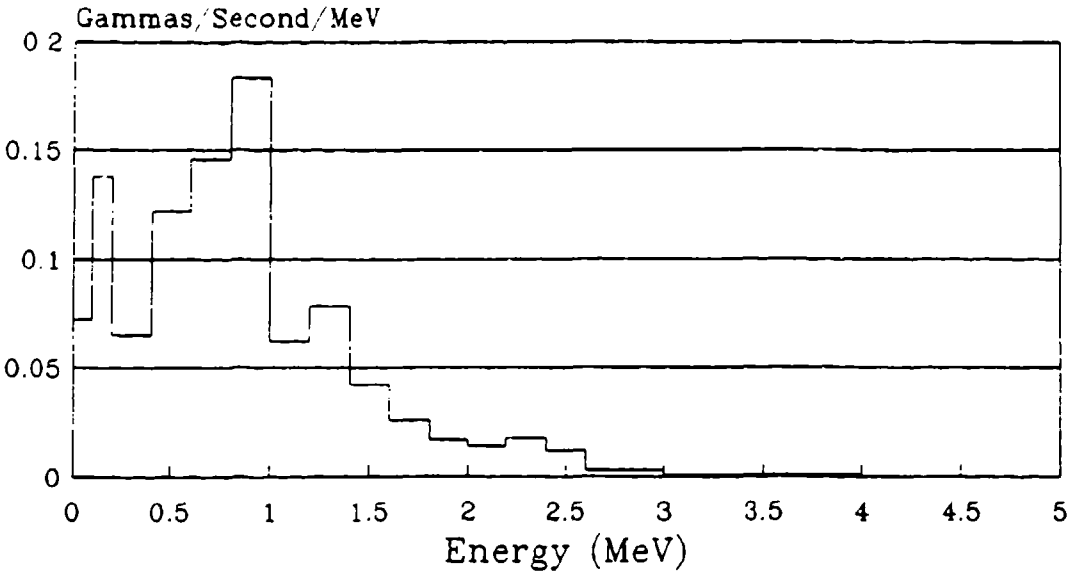


Time = 2 Hour

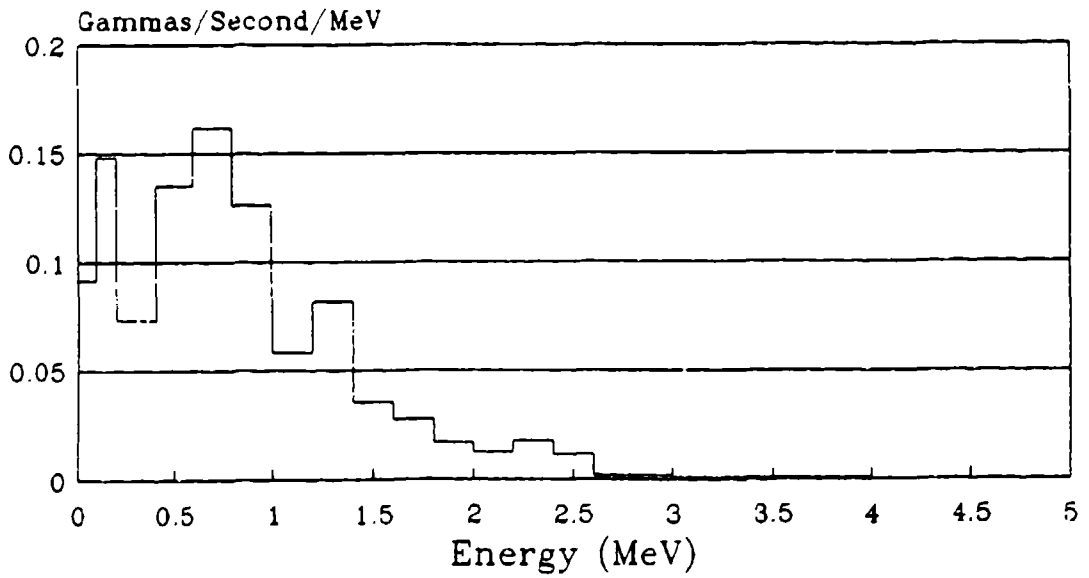
U-235



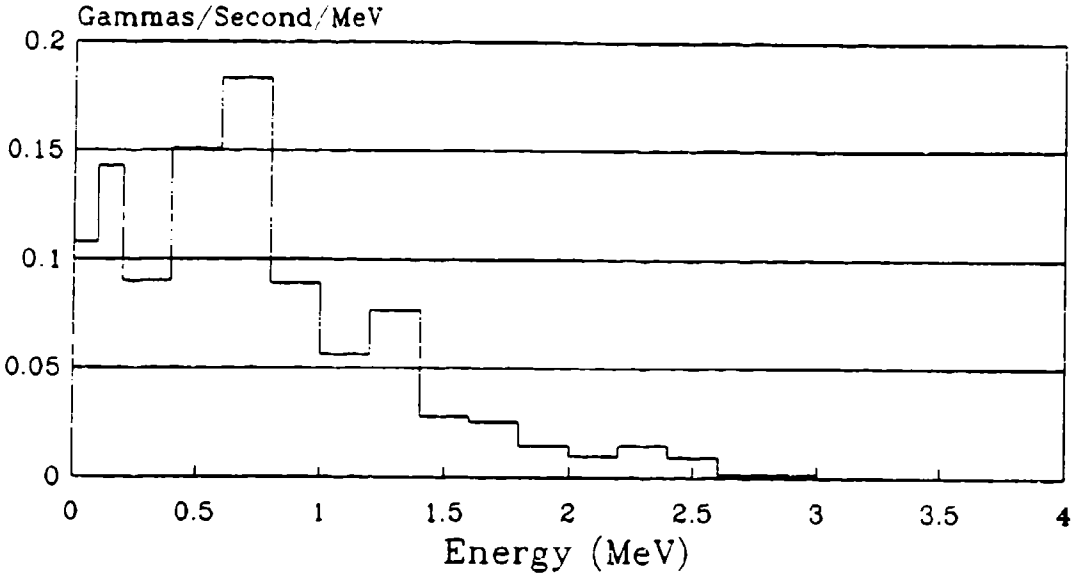
Time = 4 Hours
U-235



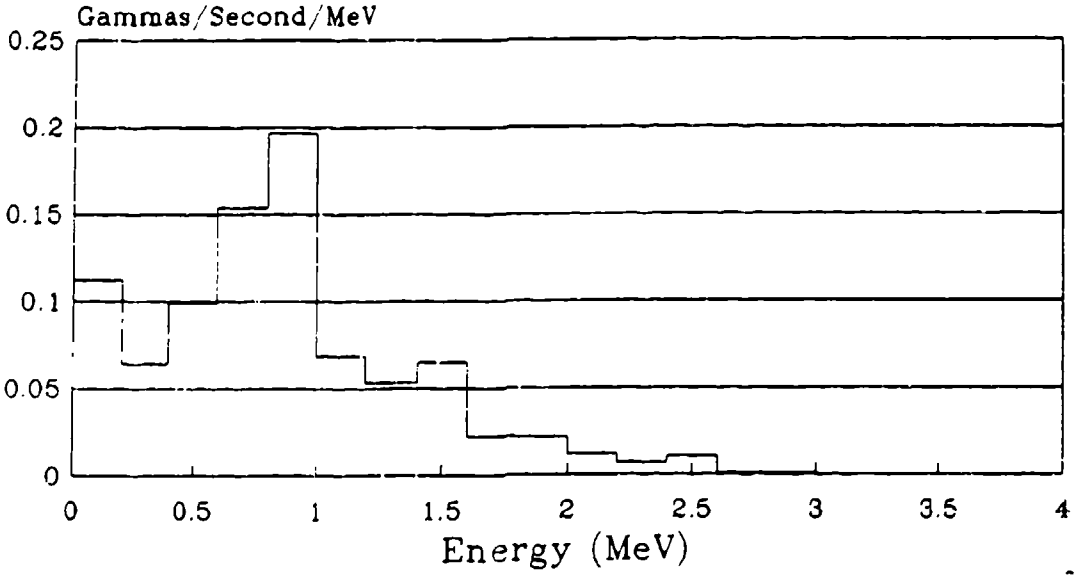
Time = 6 Hours
U-235



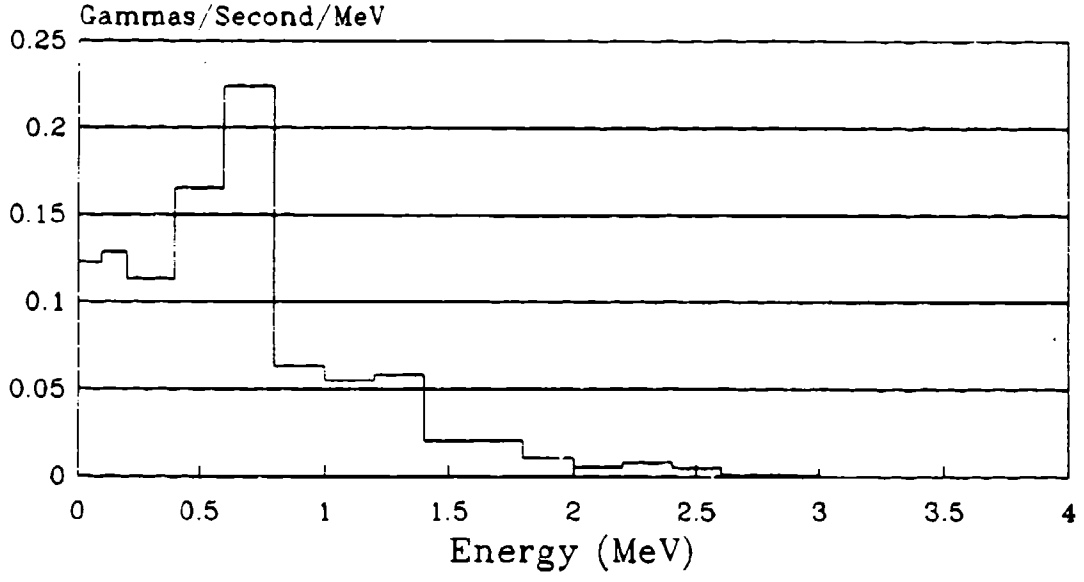
Time = 8 Hours
U-235



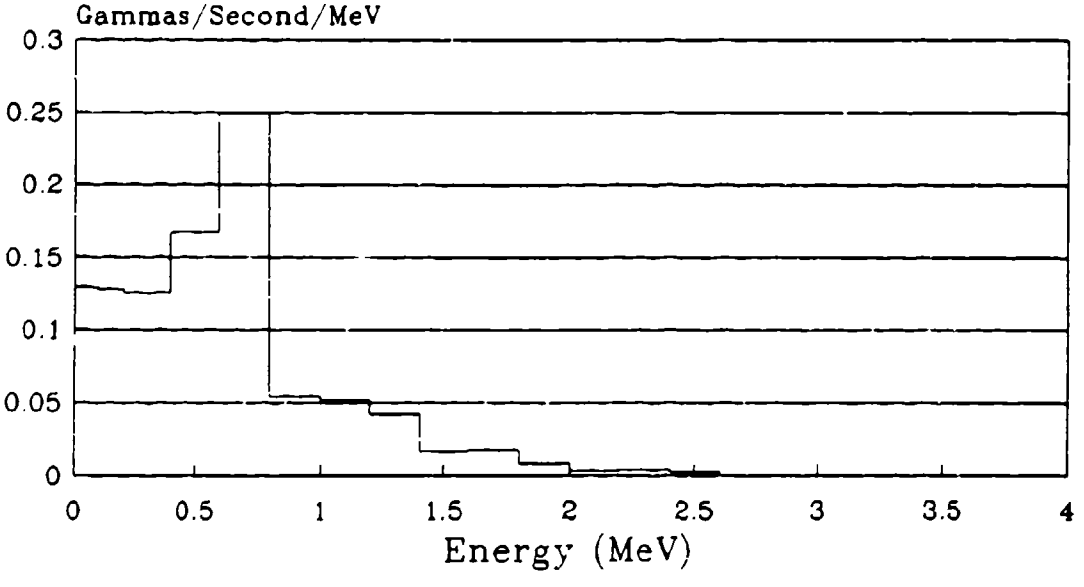
Time = 10 Hours
U-235



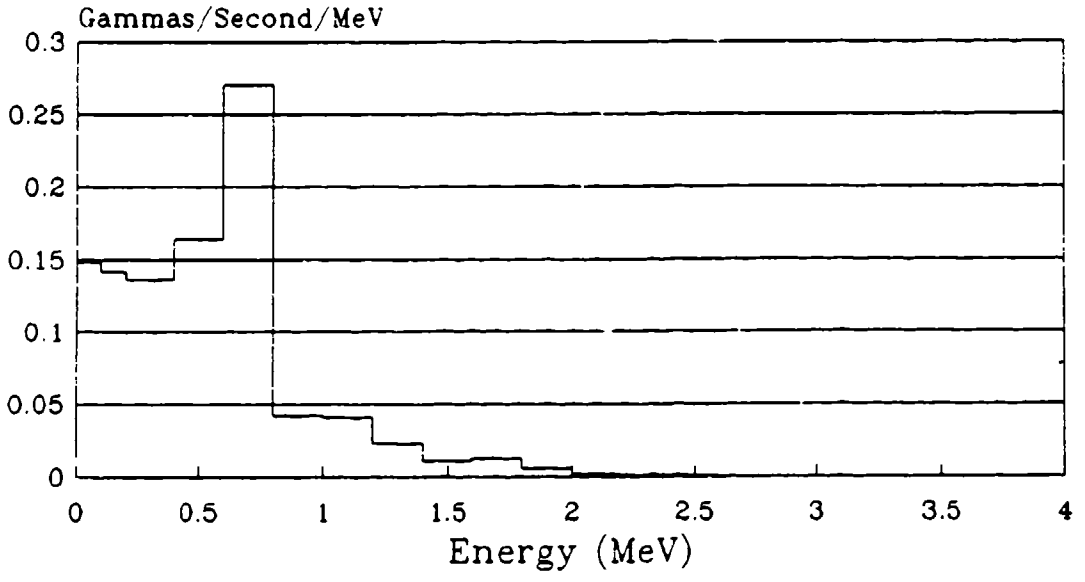
Time = 12 Hours
U-235



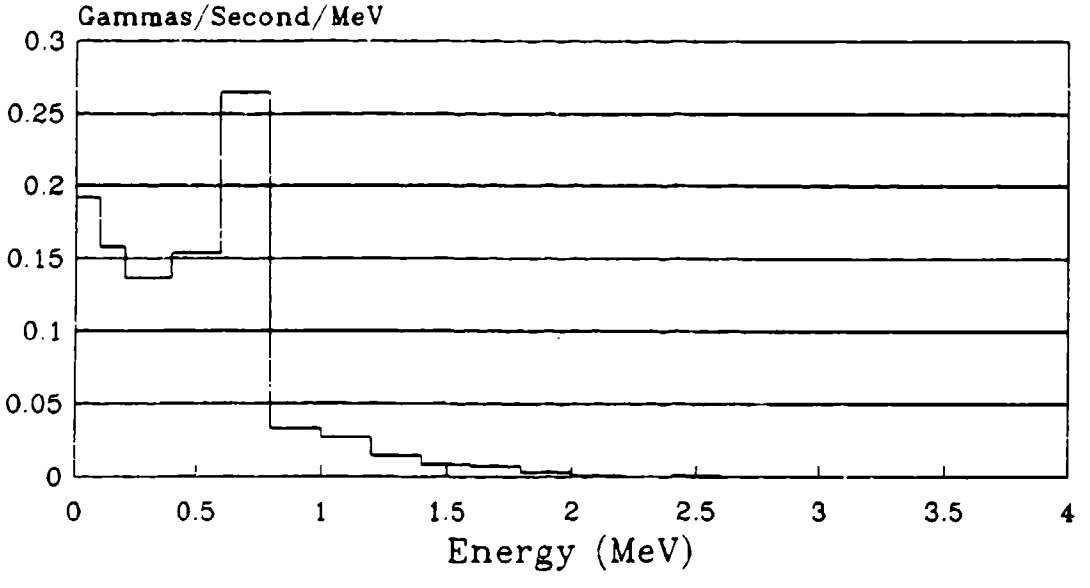
Time = 16 Hours
U-235



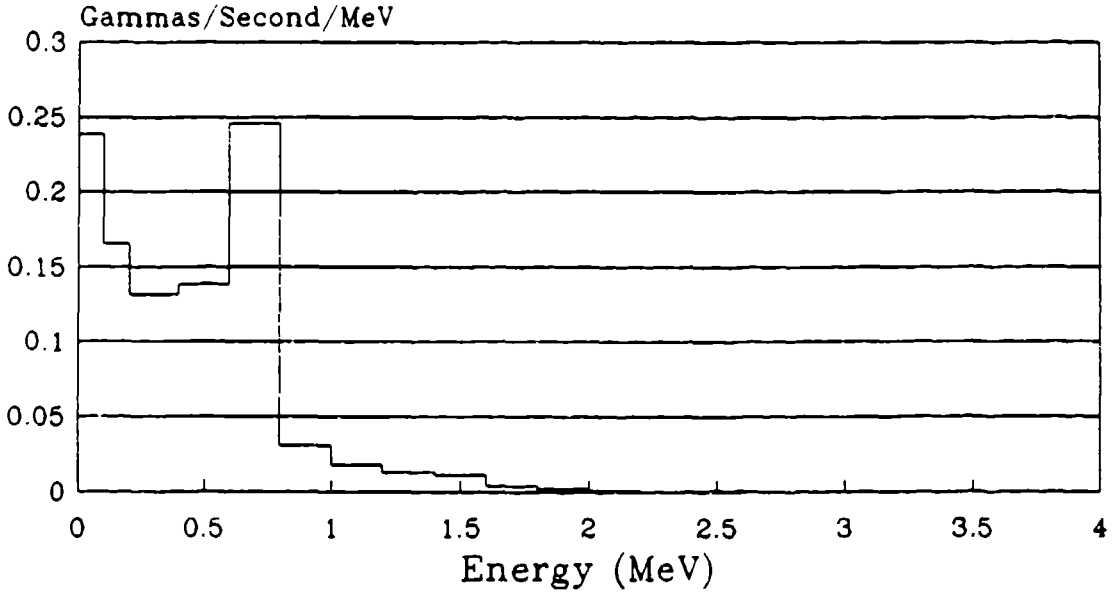
Time = 24 Hours
U-235



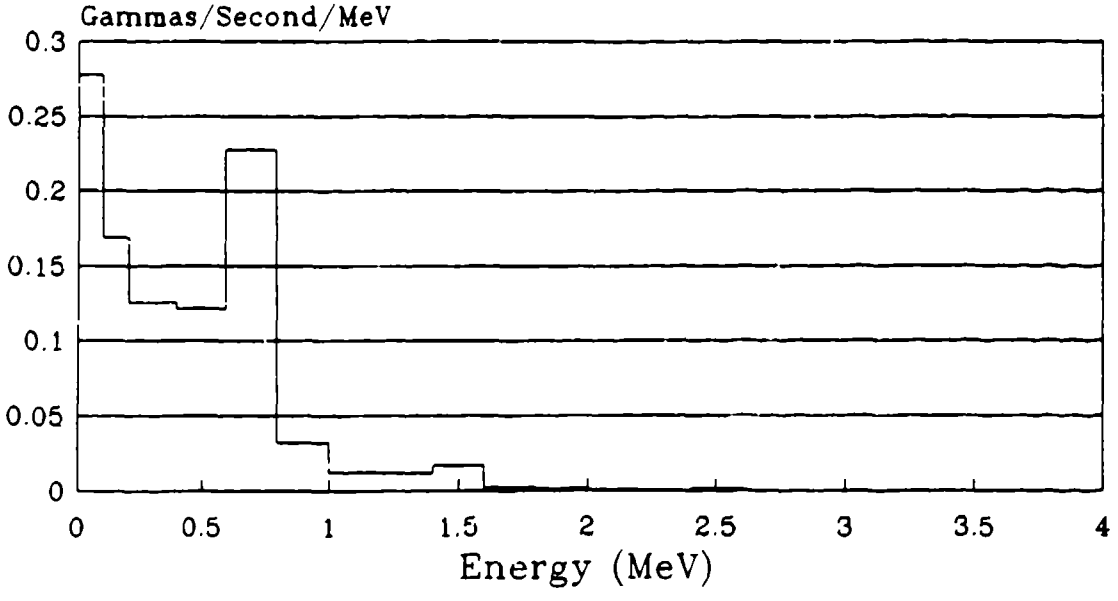
Time = 36 Hours
U-235



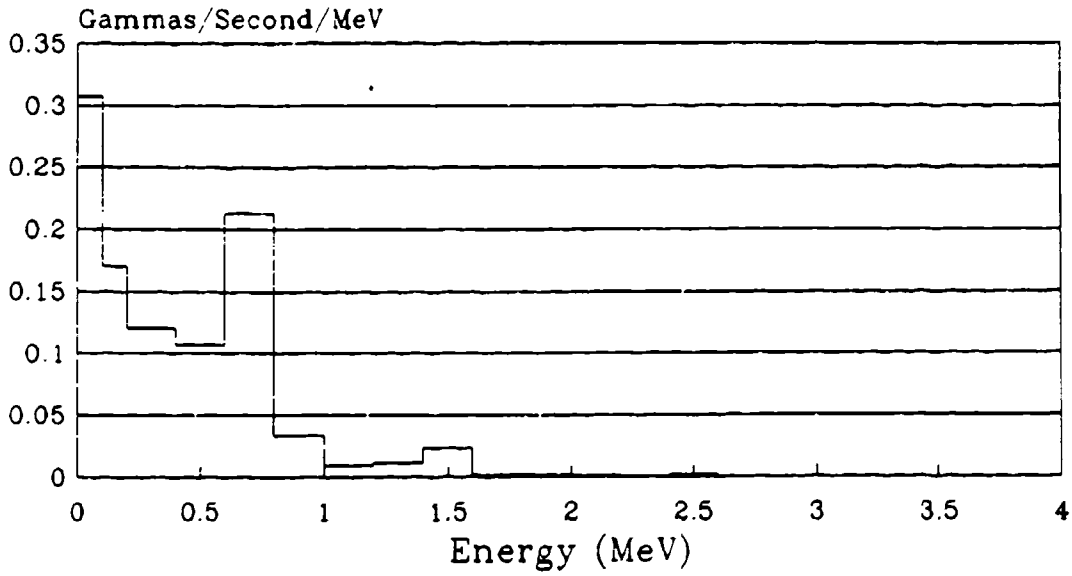
Time = 48 Hours
U-235



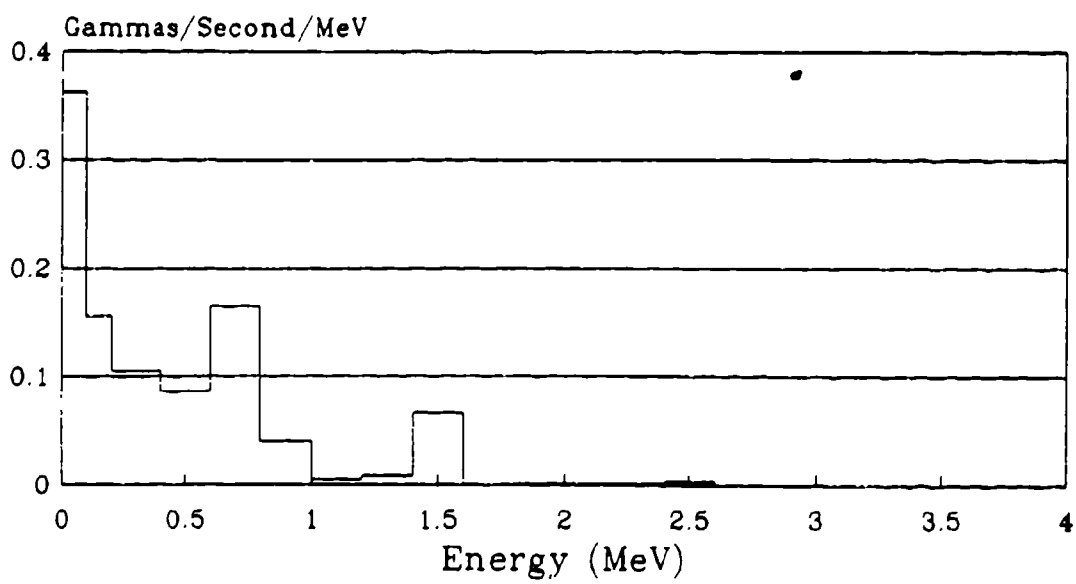
Time = 60 Hours
U-235



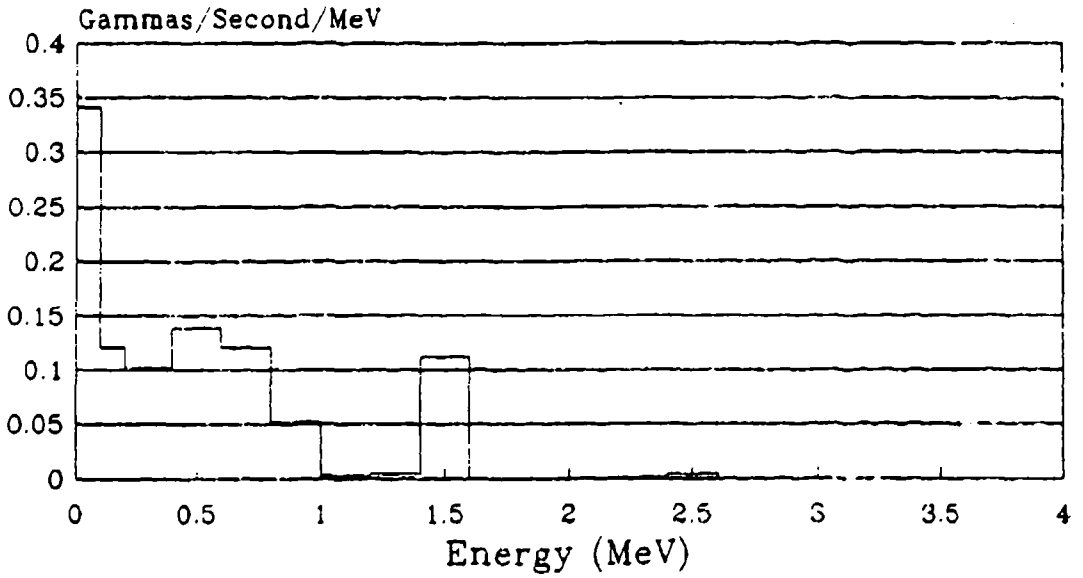
Time = 72 Hours
U-235



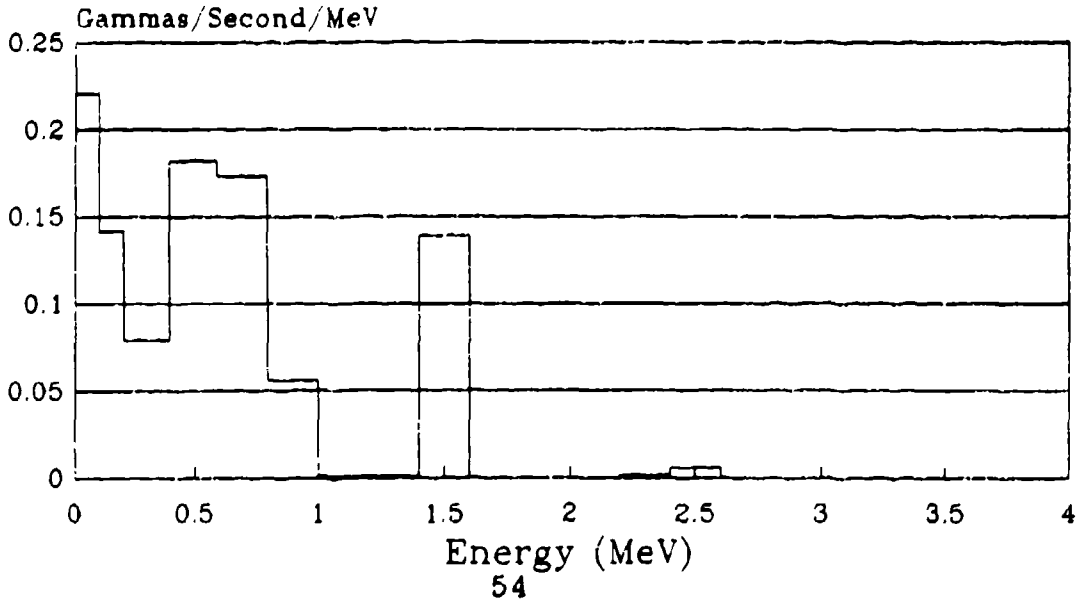
Time = 7 Days
U-235



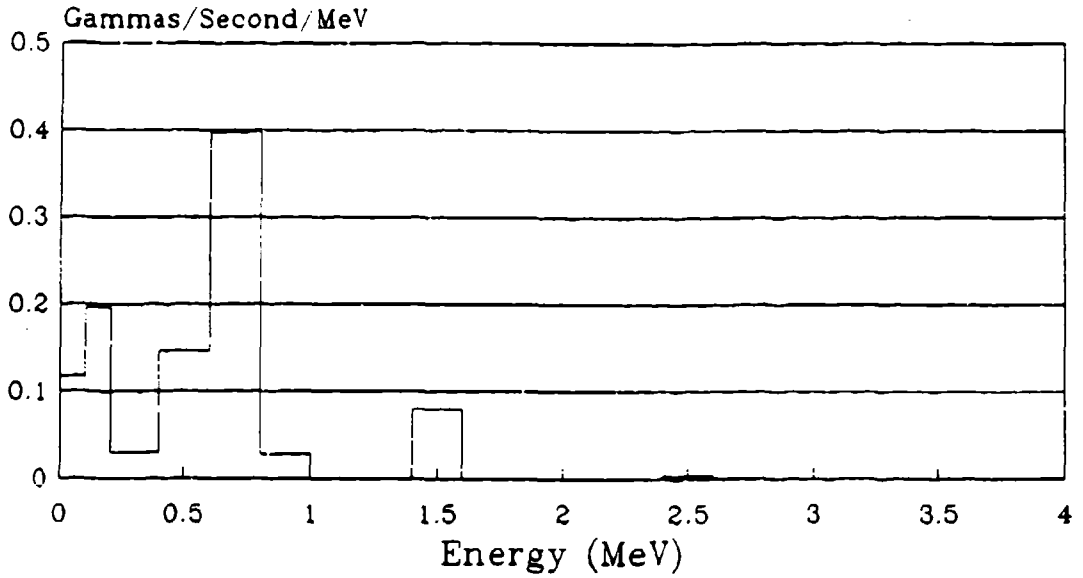
Time = 14 Days
U-235



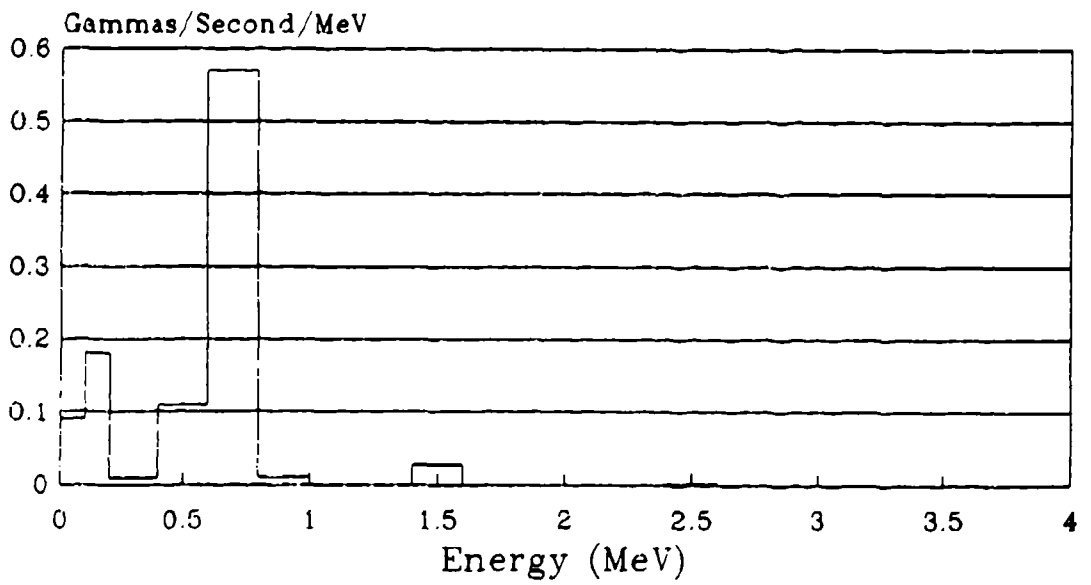
Time = 30 Days
U-235



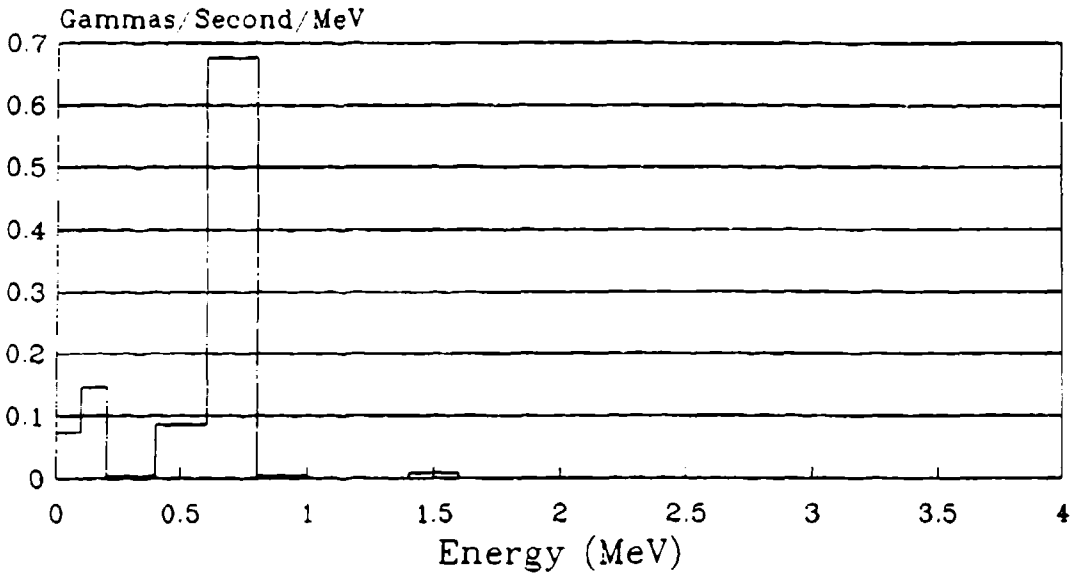
Time = 60 Days
U-235



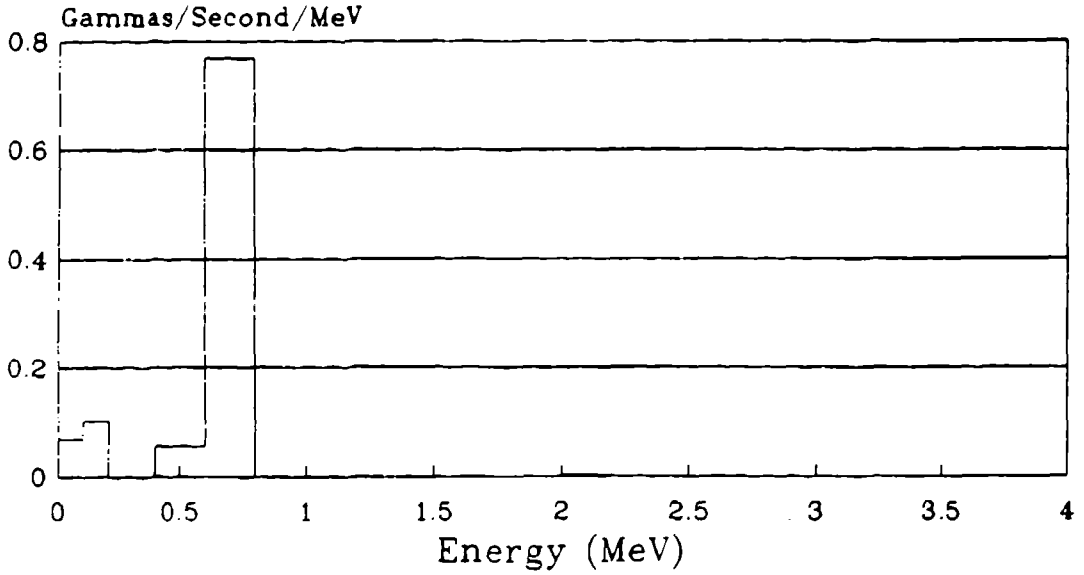
Time = 90 Days
U-235



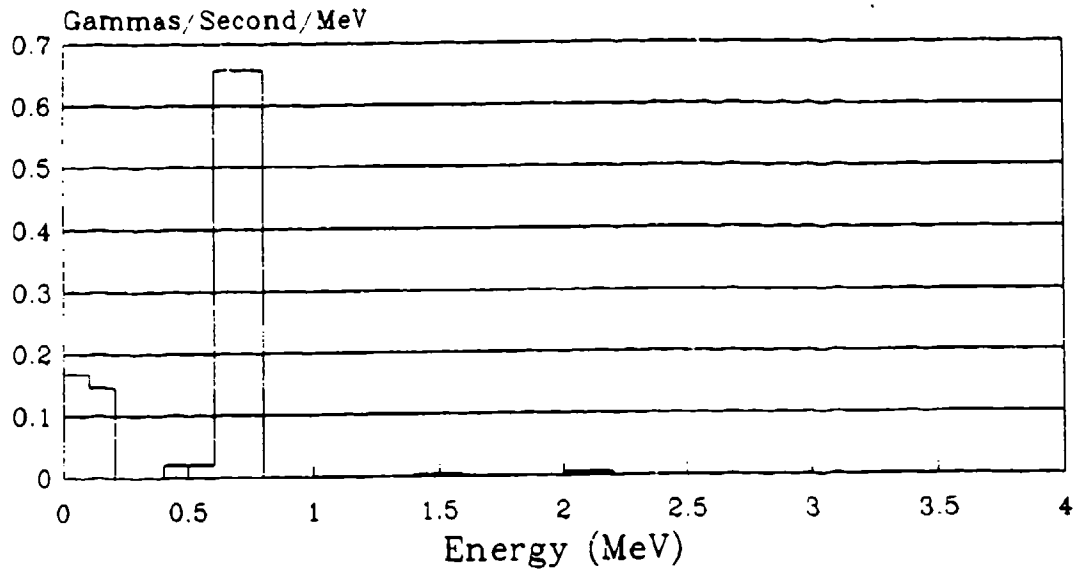
Time = 120 Days
U-235



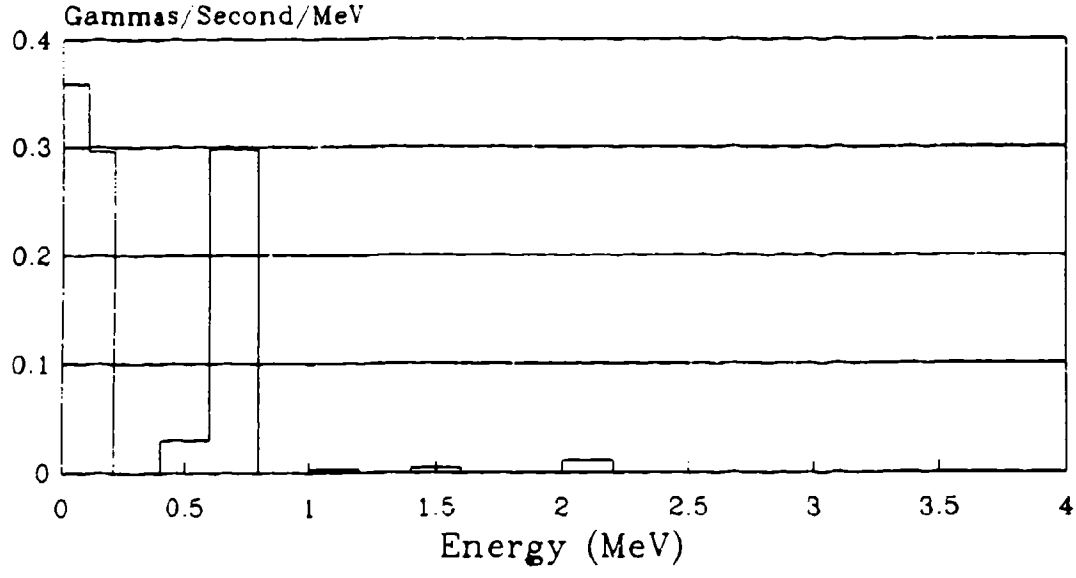
Time = 180 Days
U-235



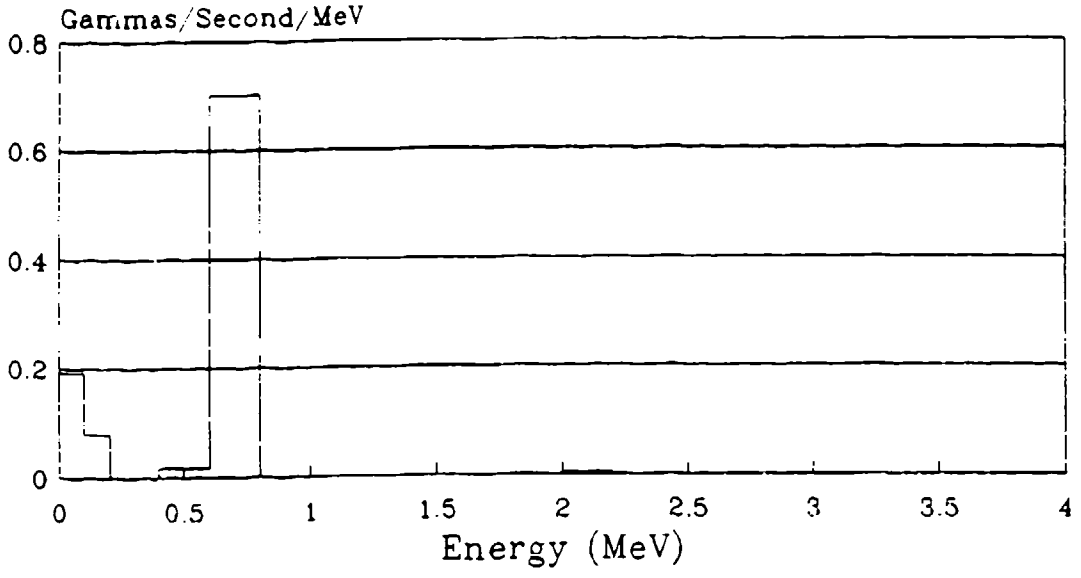
Time = 1 Year
U-235



Time = 2 Years
U-235



Time = 5 Years
U-235

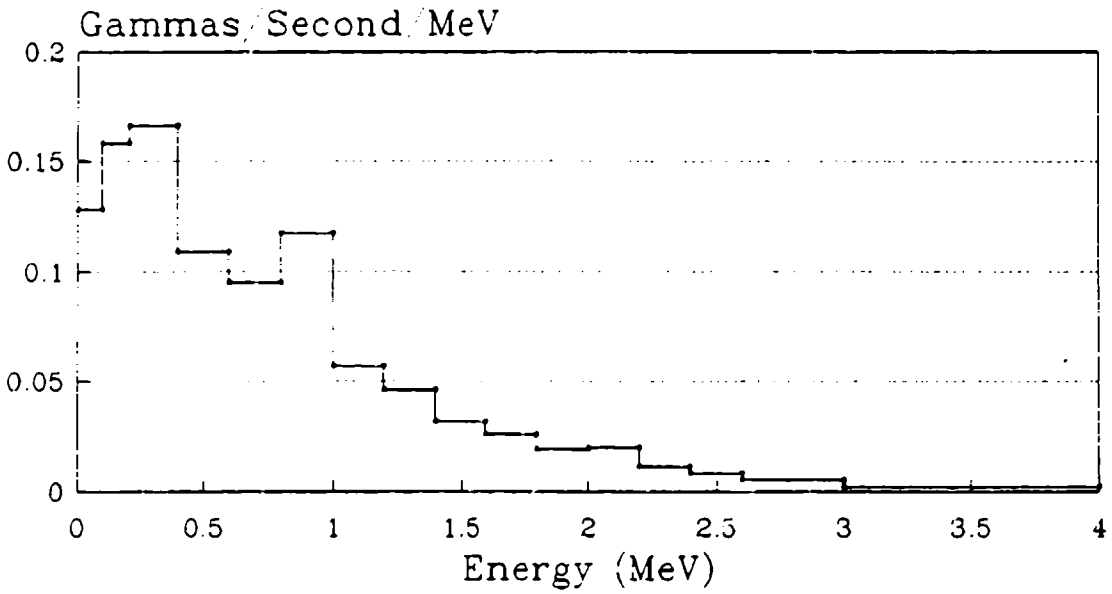


Appendix B

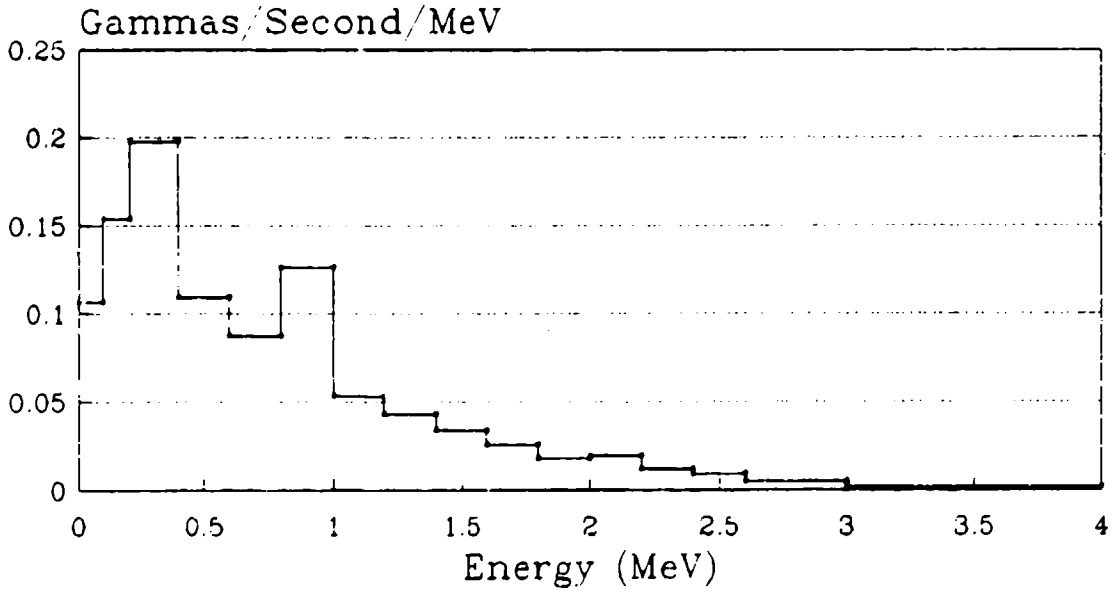
Pu-239 Fission Product Gamma Ray Spectra From 10 Minutes to 5 Years

The following spectra are from the fission products from Pu-239 fuel. The spectra have been normalized to one gamma-ray per second per MeV.

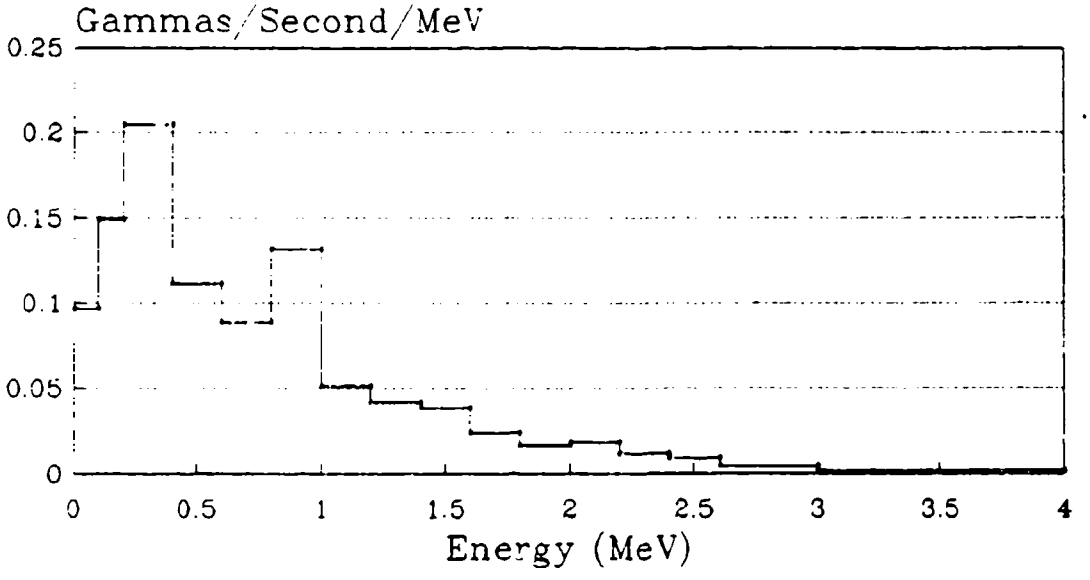
Time = 10 Minutes
Pu-239



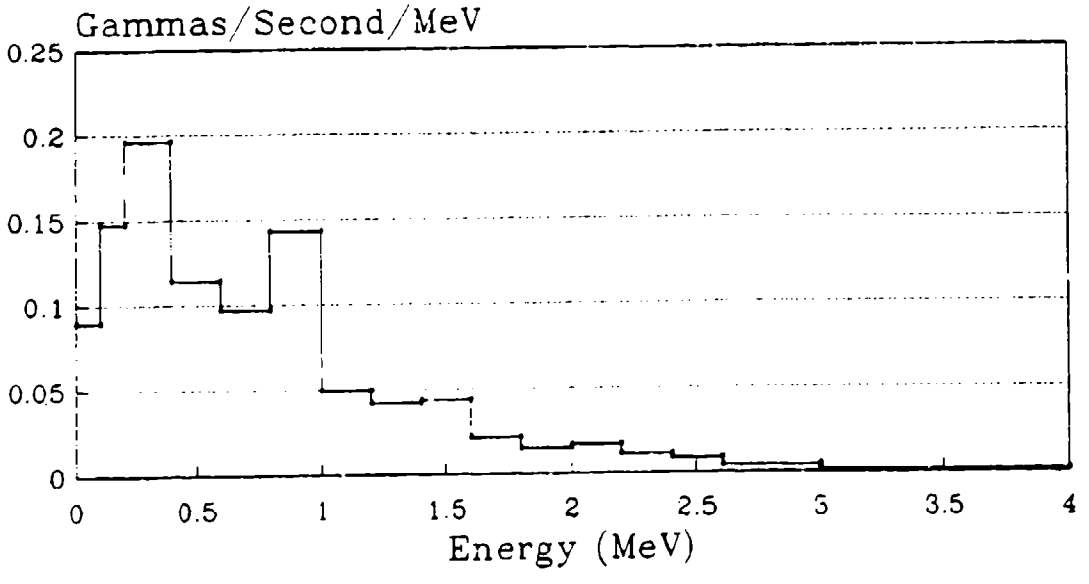
Time = 20 Minutes
Pu-239



Time = 30 Minutes
Pu-239

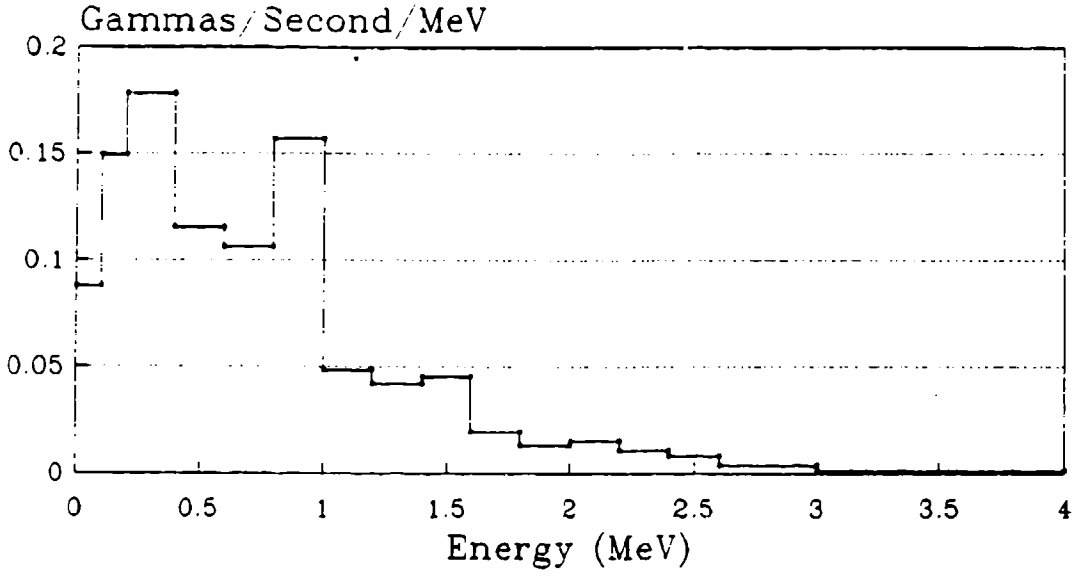


Time = 45 Minutes
Pu-239



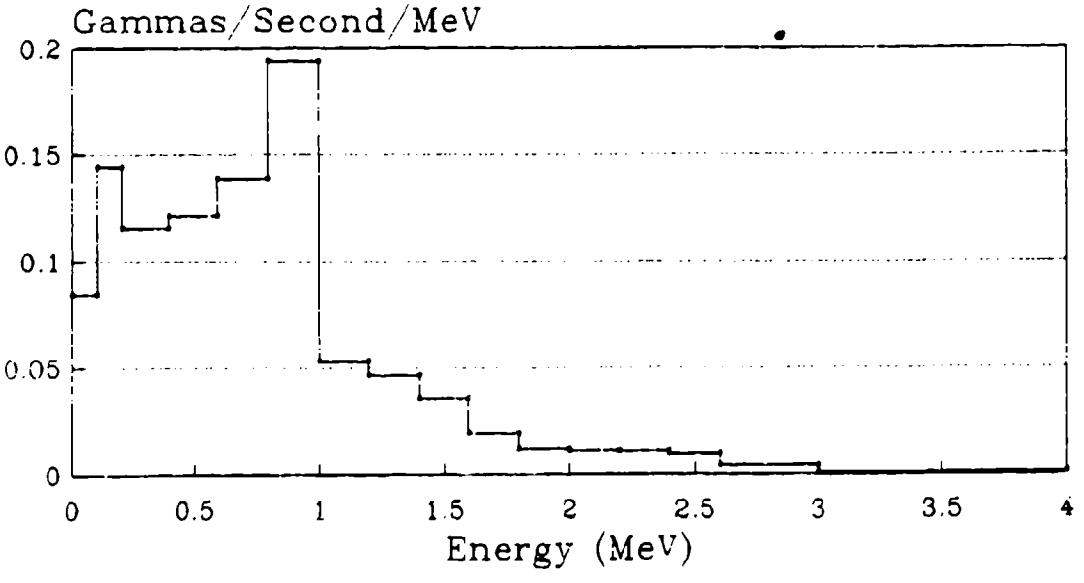
Time = 1 Hour

Pu-239

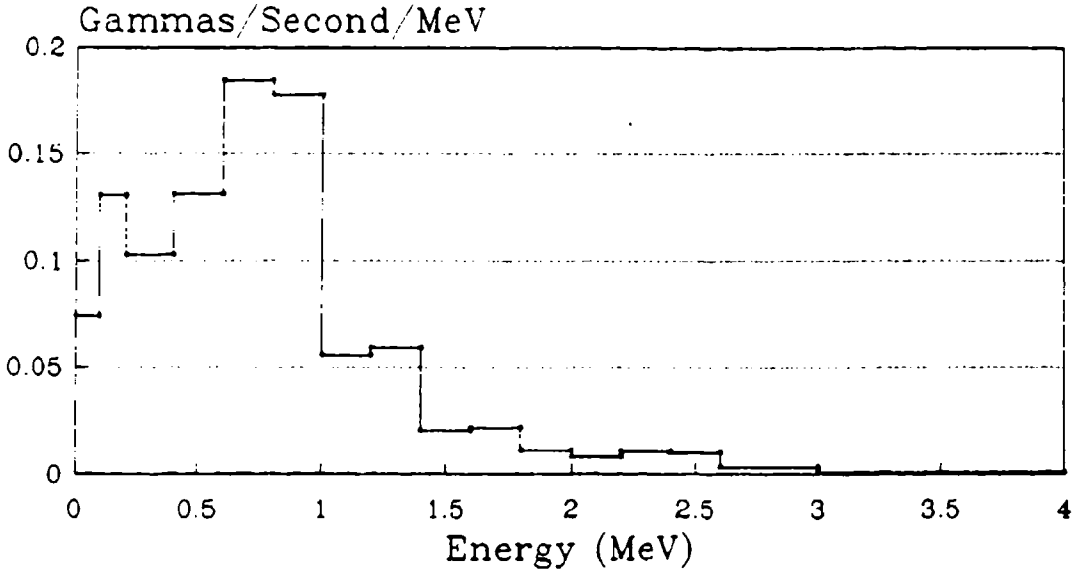


Time = 2 Hours

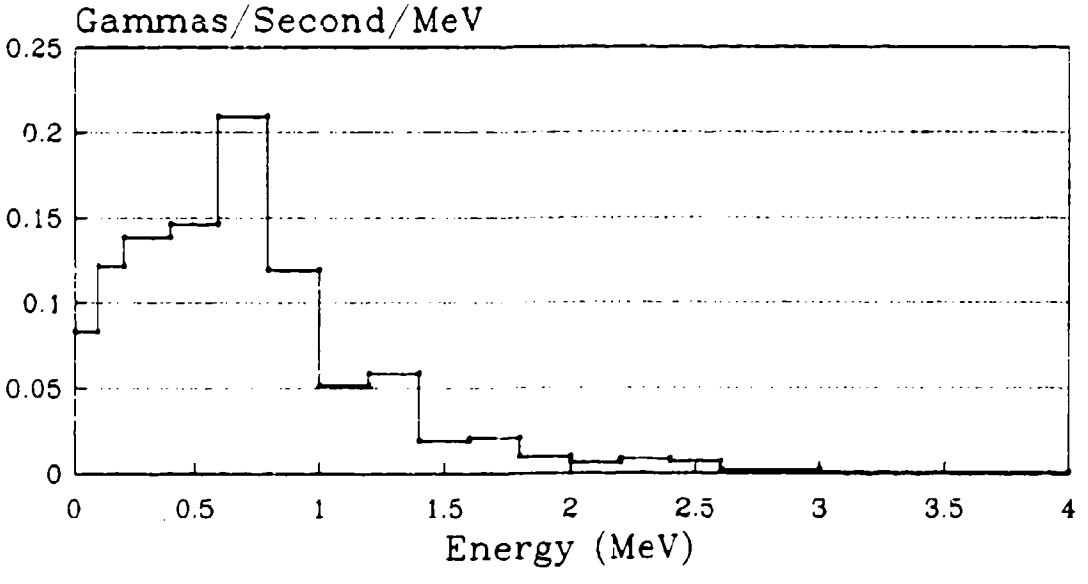
Pu-239



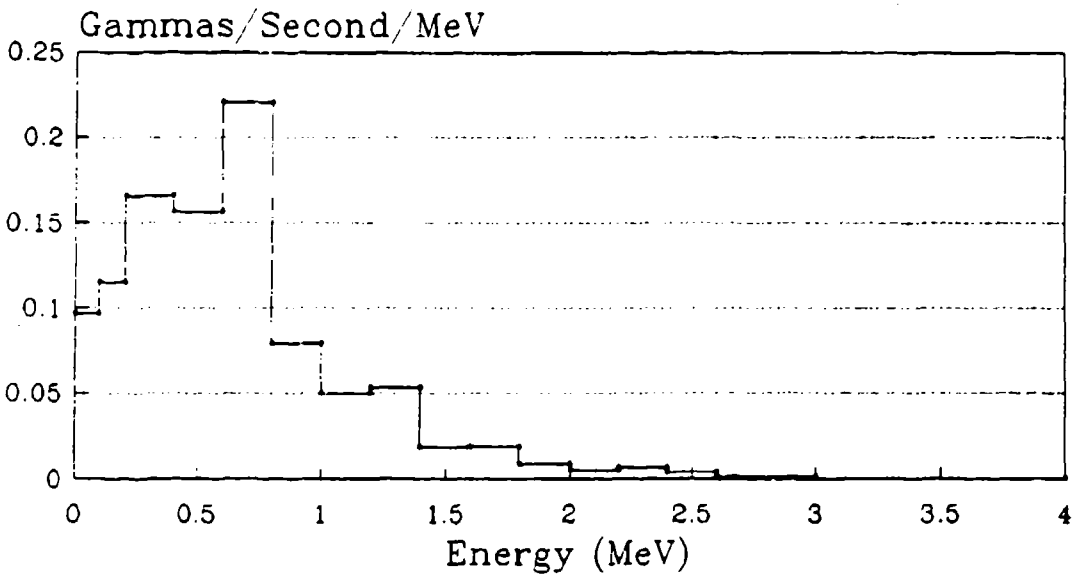
Time = 4 Hours
Pu-239



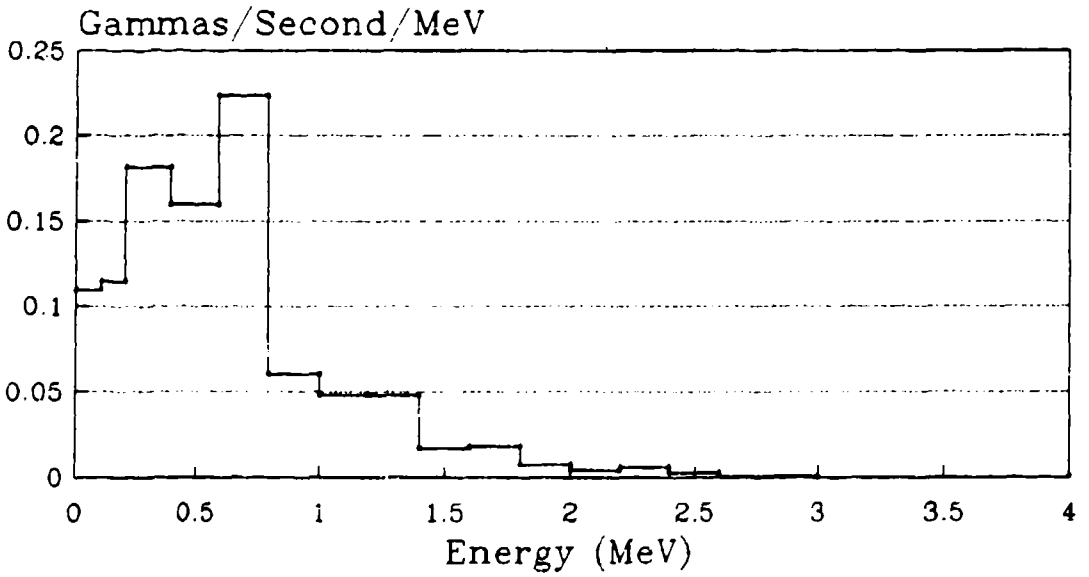
Time = 6 Hours
Pu-239



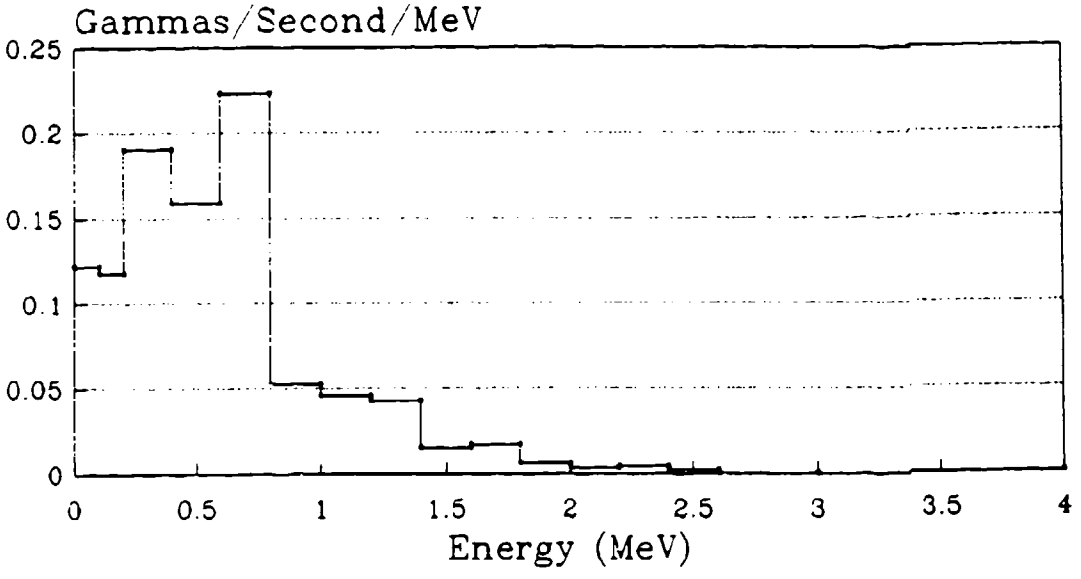
Time = 8 Hours
Pu-239



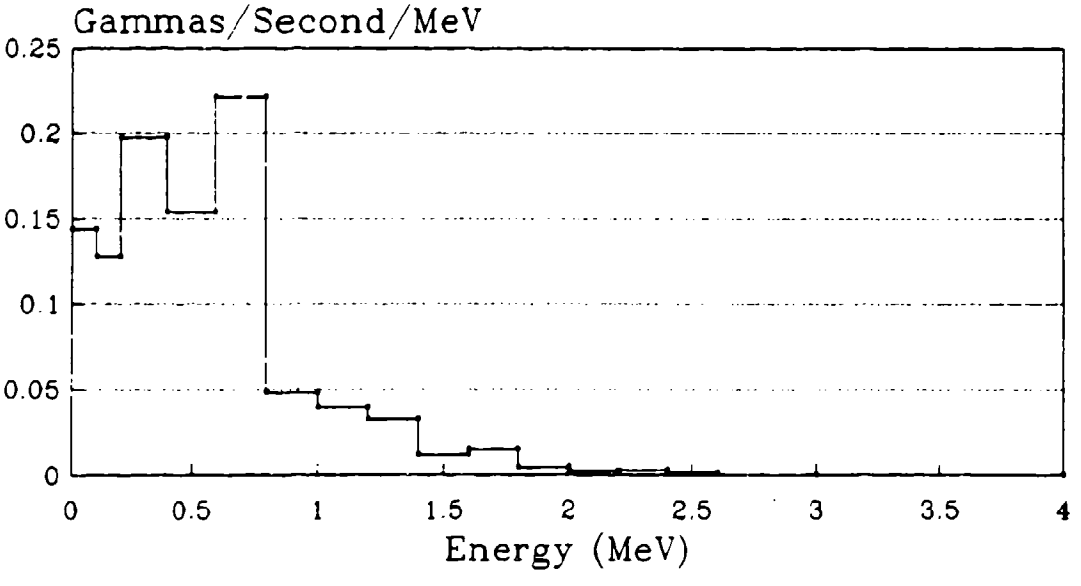
Time = 10 Hours
Pu-239



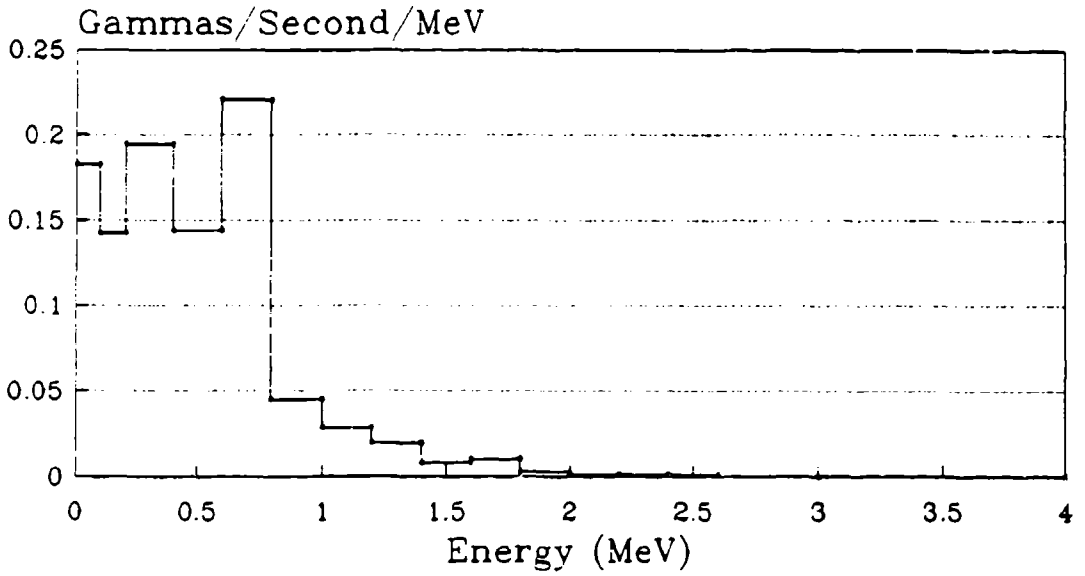
Time = 12 Hours
Pu-239



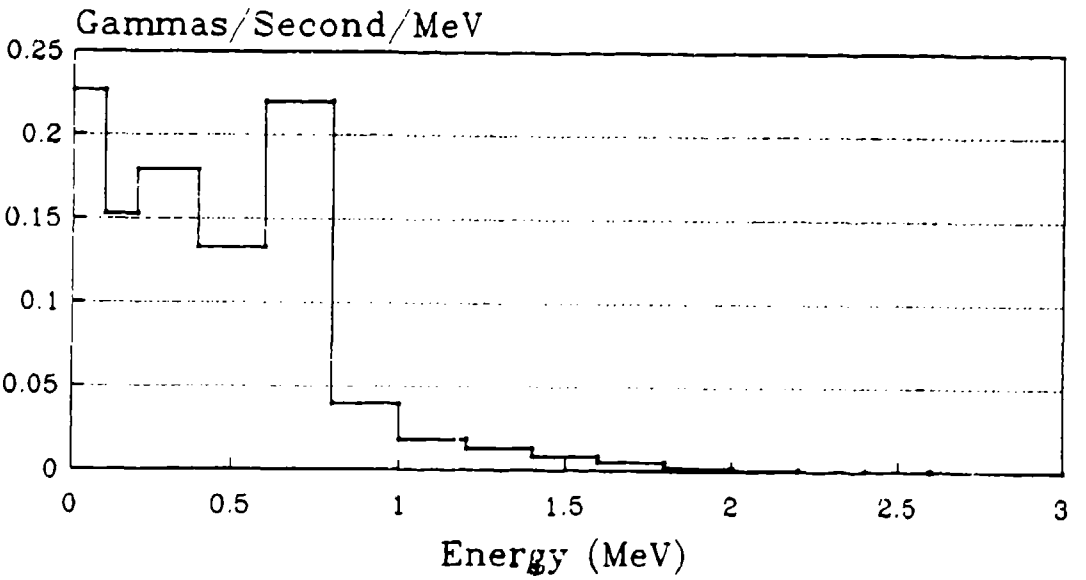
Time = 16 Hours
Pu-239



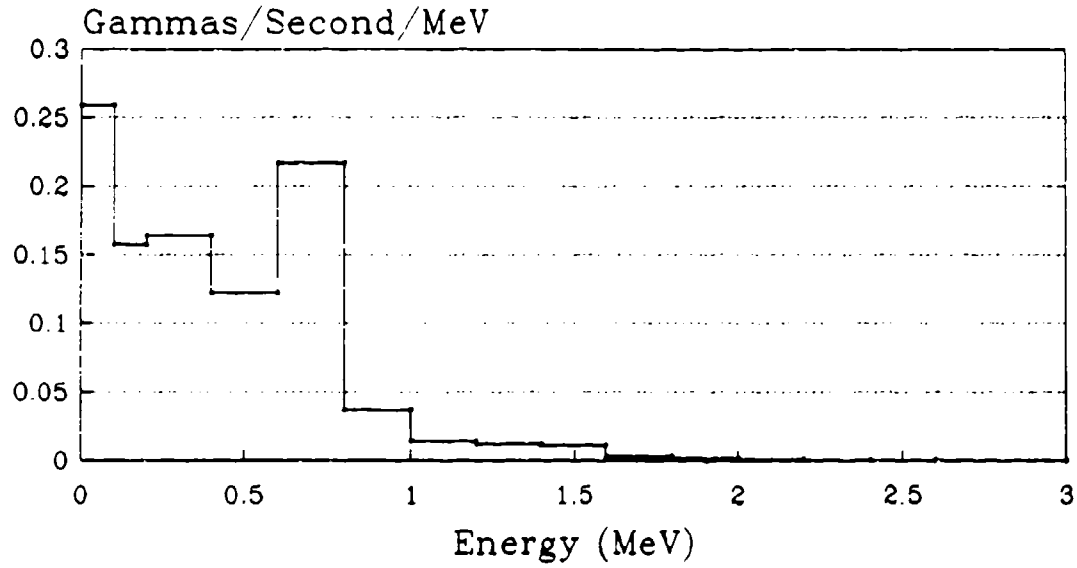
Time = 24 Hours
Pu-239



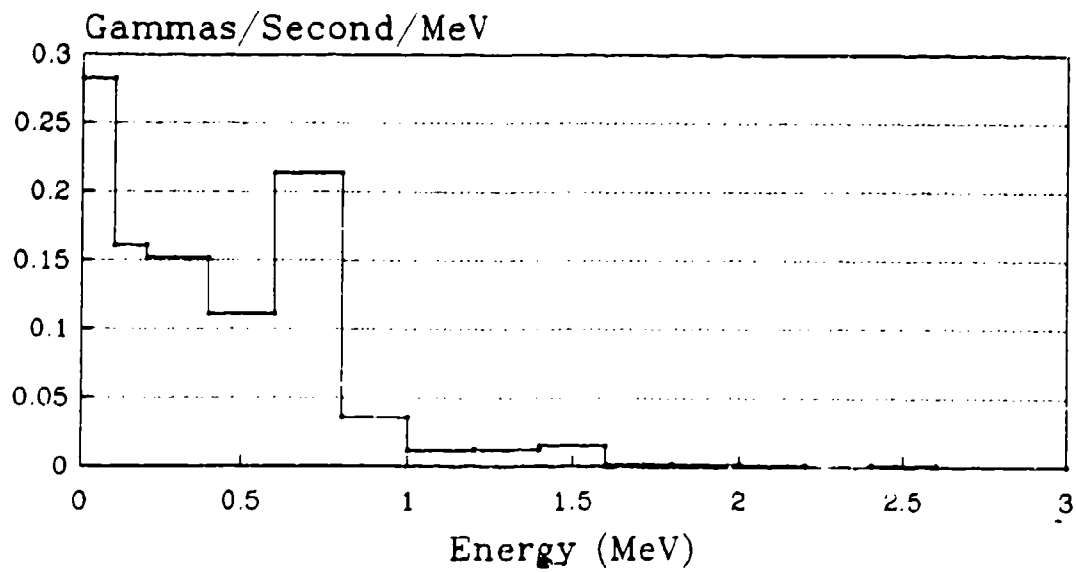
Time = 36 Hours
Pu-239



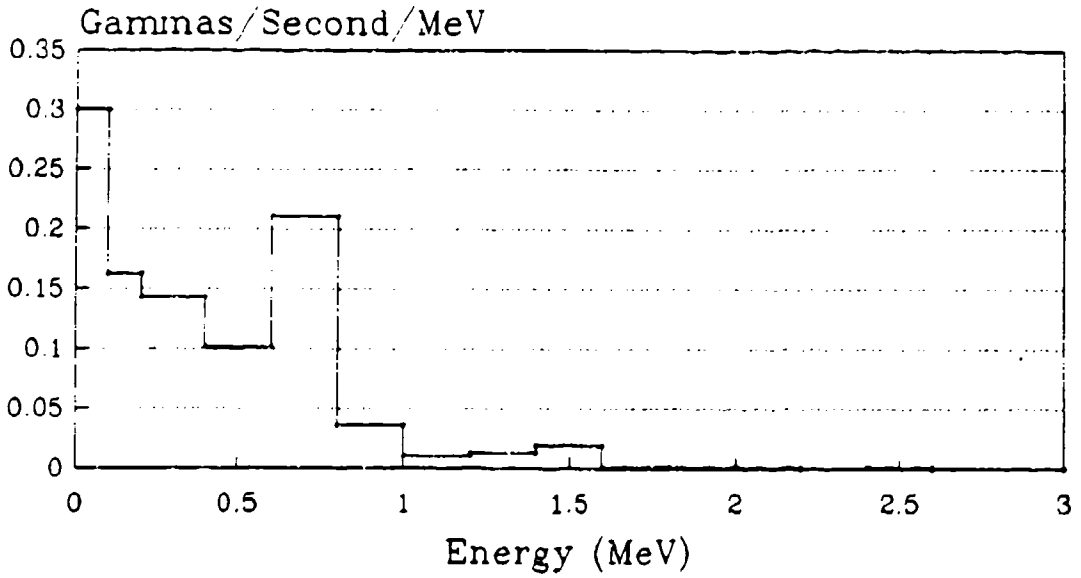
Time = 48 Hours
Pu-239



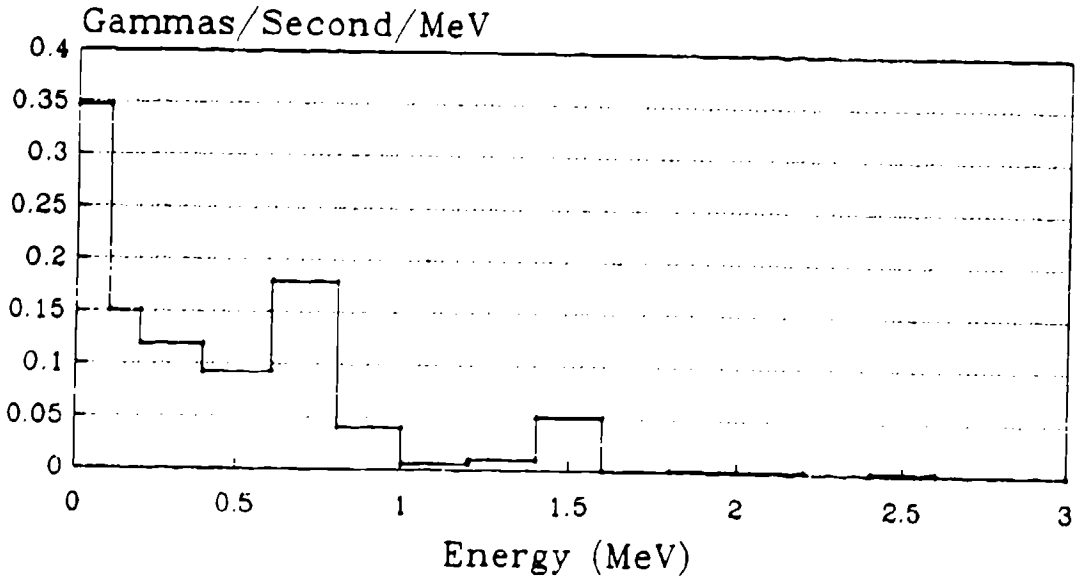
Time = 60 Hours
Pu-239



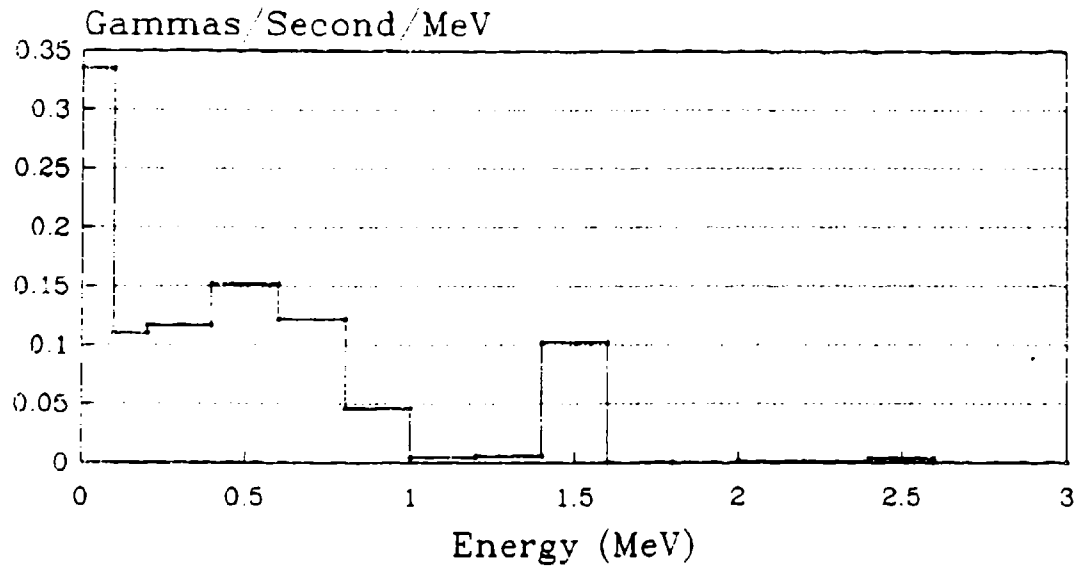
Time = 72 Hours
Pu-239



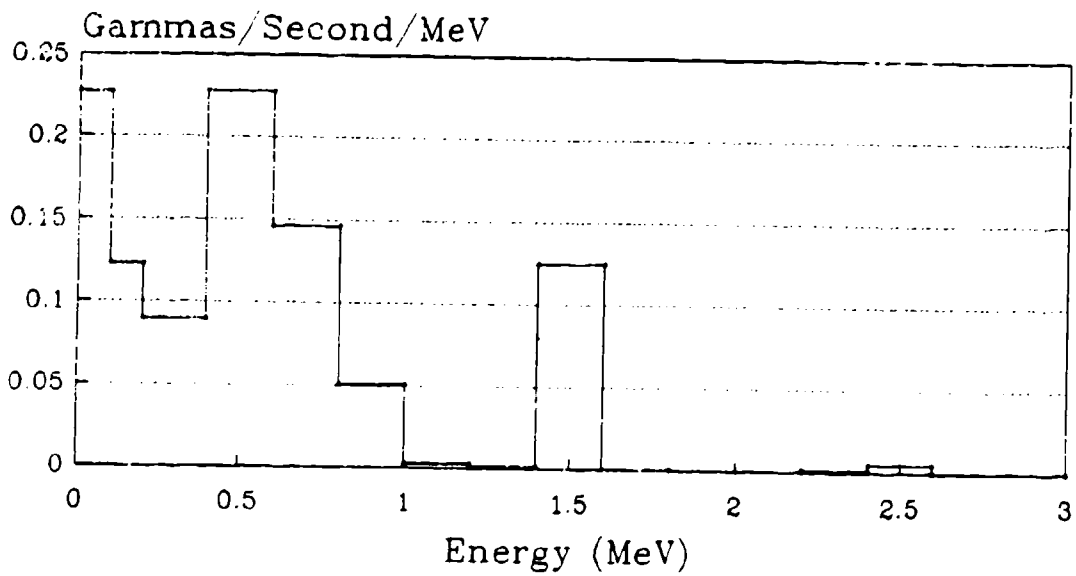
Time = 7 Days
Pu-239



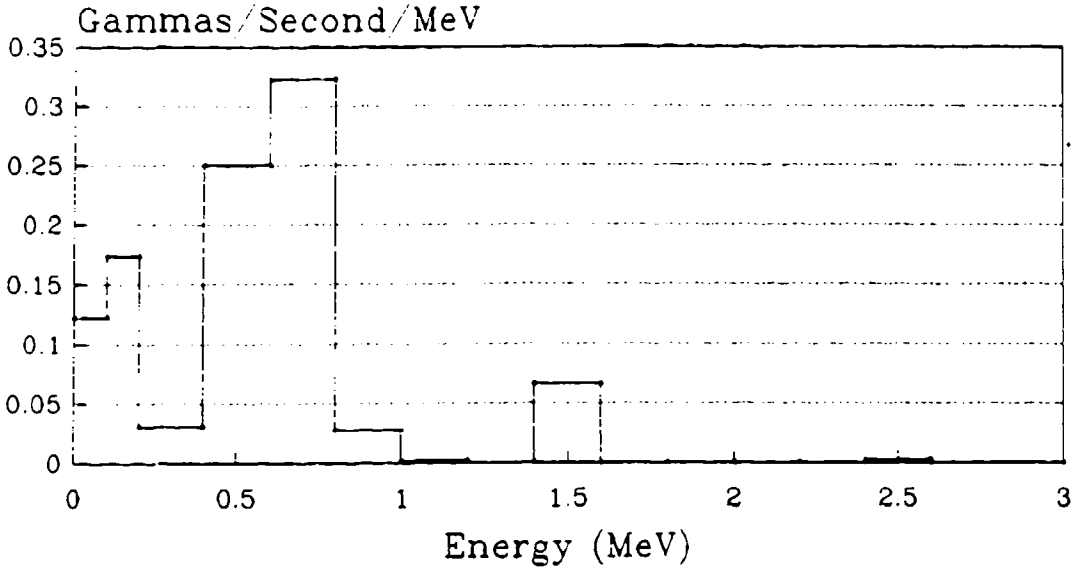
Time = 14 Days
Pu-239



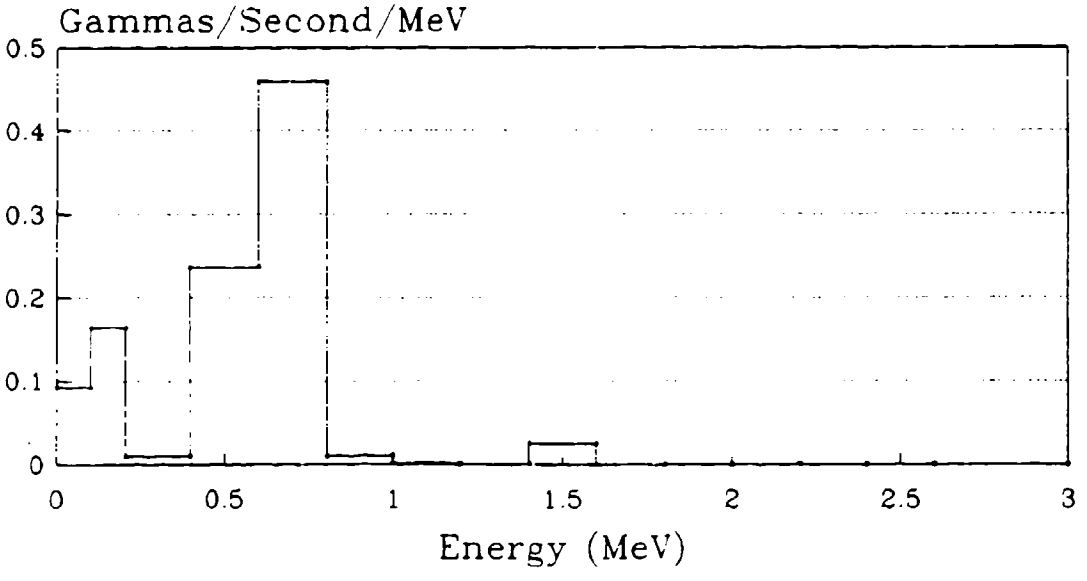
Time = 30 Days
Pu-239



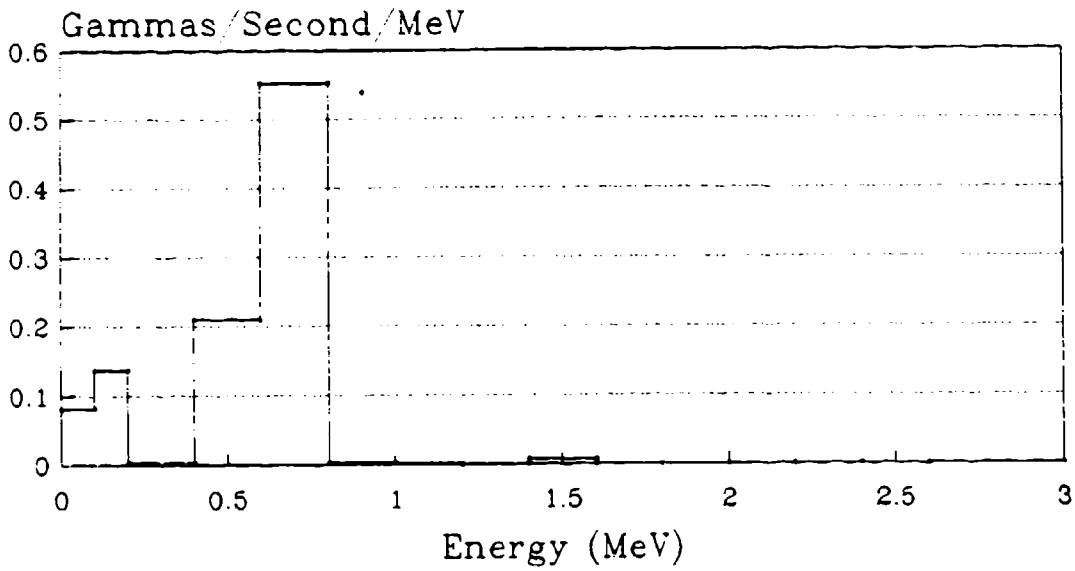
Time = 60 Days
Pu-239



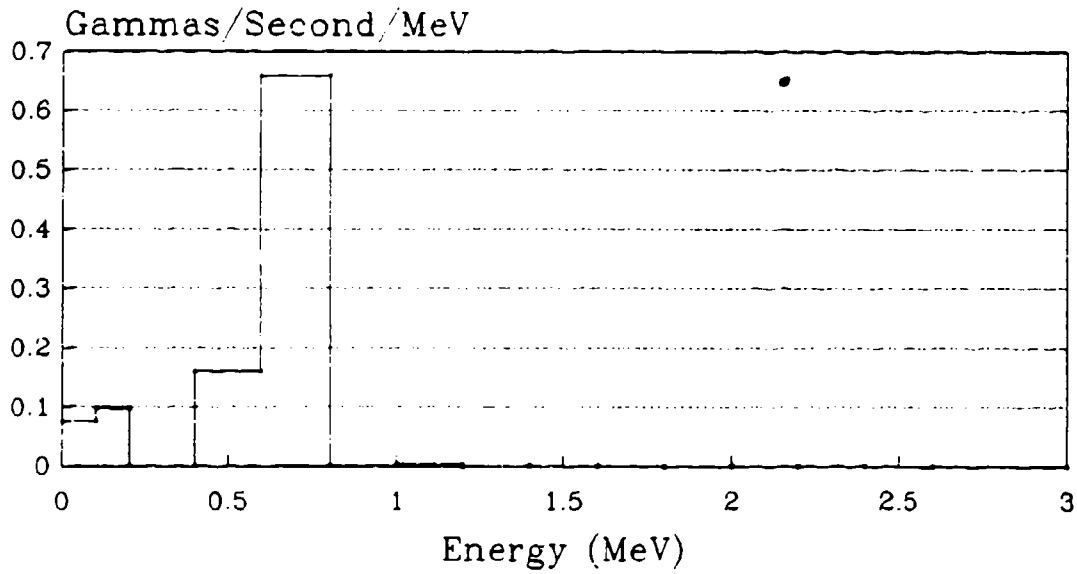
Time = 90 Days
Pu-239



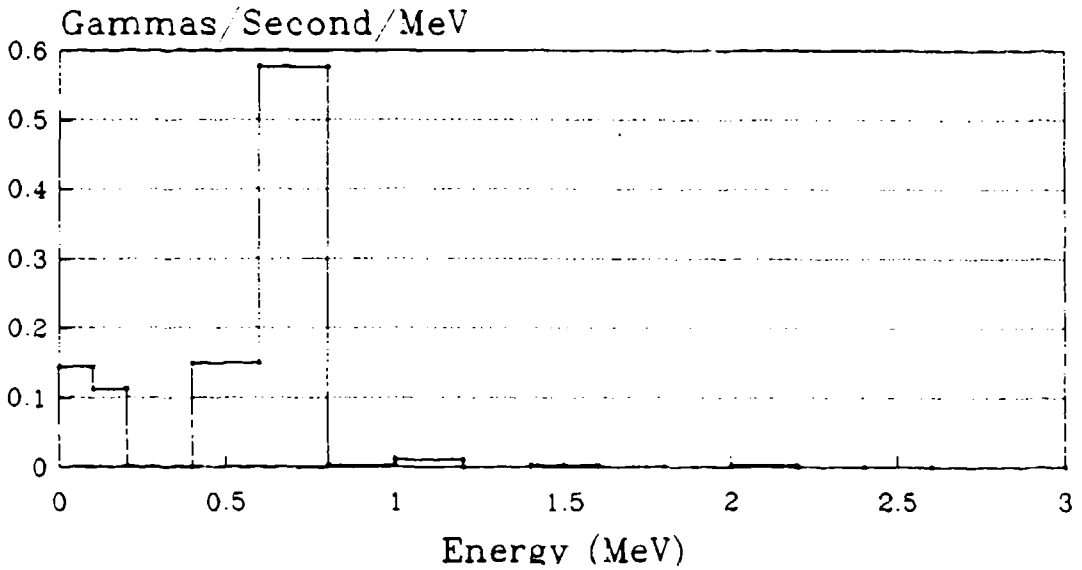
Time = 4 Months
Pu-239



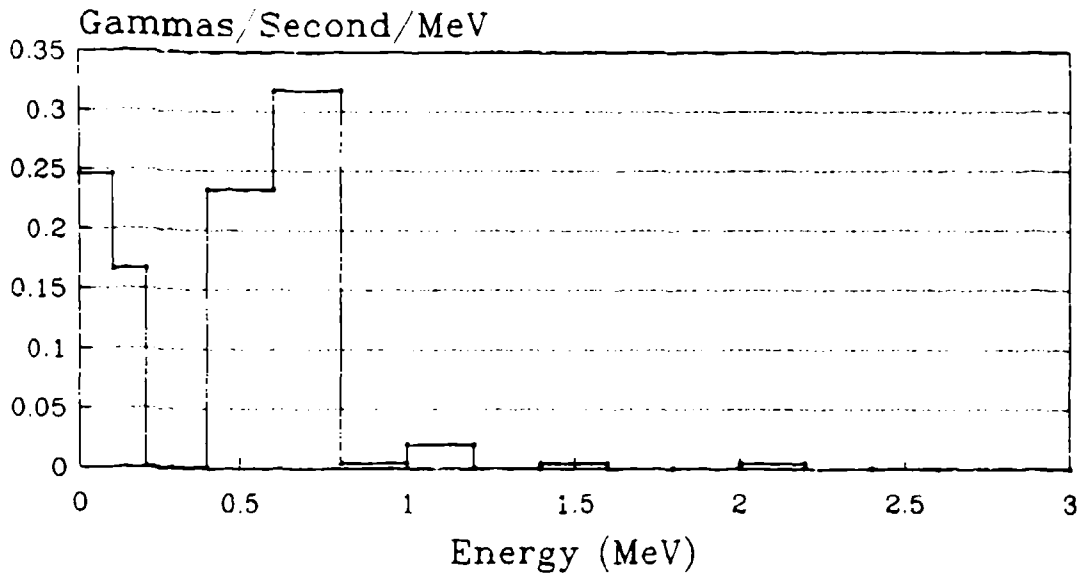
Time = 6 Months
Pu-239



Time = 1 Year
Pu-239

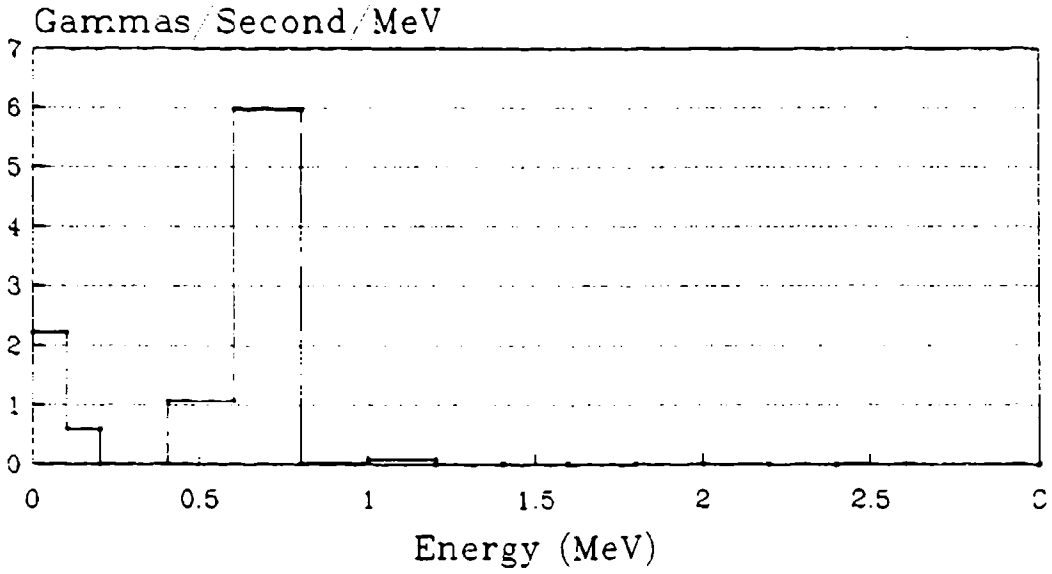


Time = 2 Years
Pu-239



Time = 5 Years

Pu-239

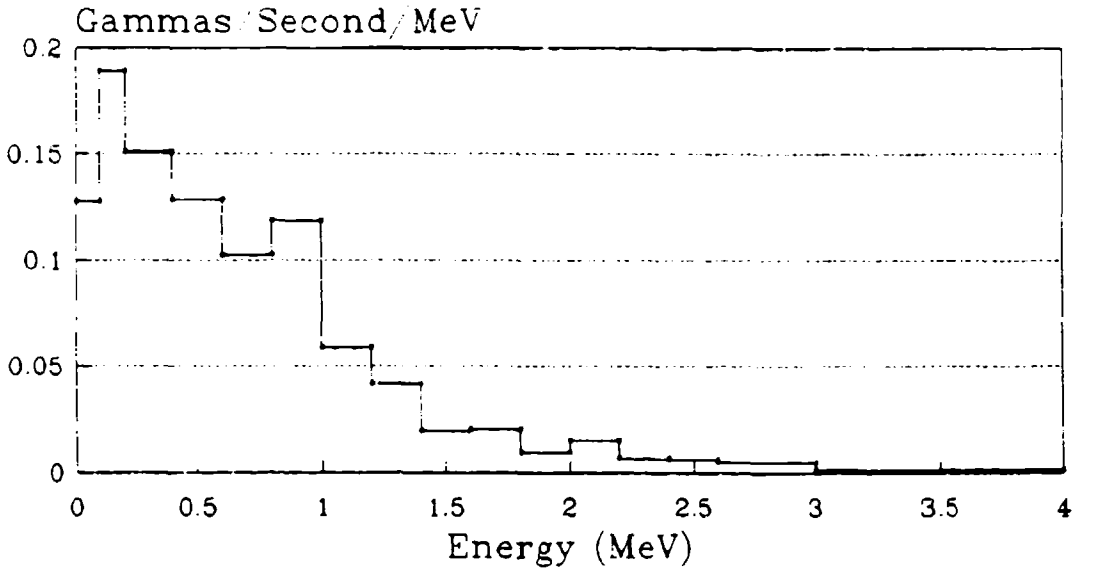


Appendix C

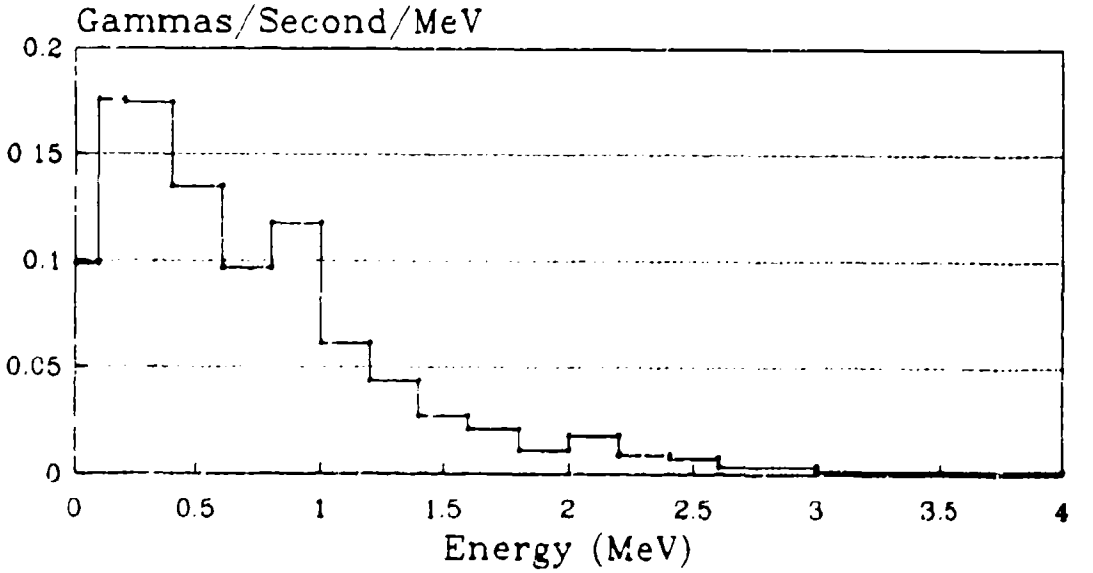
U-238 Fission Product Gamma Ray Spectra From 10 Minutes to 5 Years

The following spectra are from the fission products from U-238 fuel. The spectra have been normalized to one gamma-ray per second per MeV.

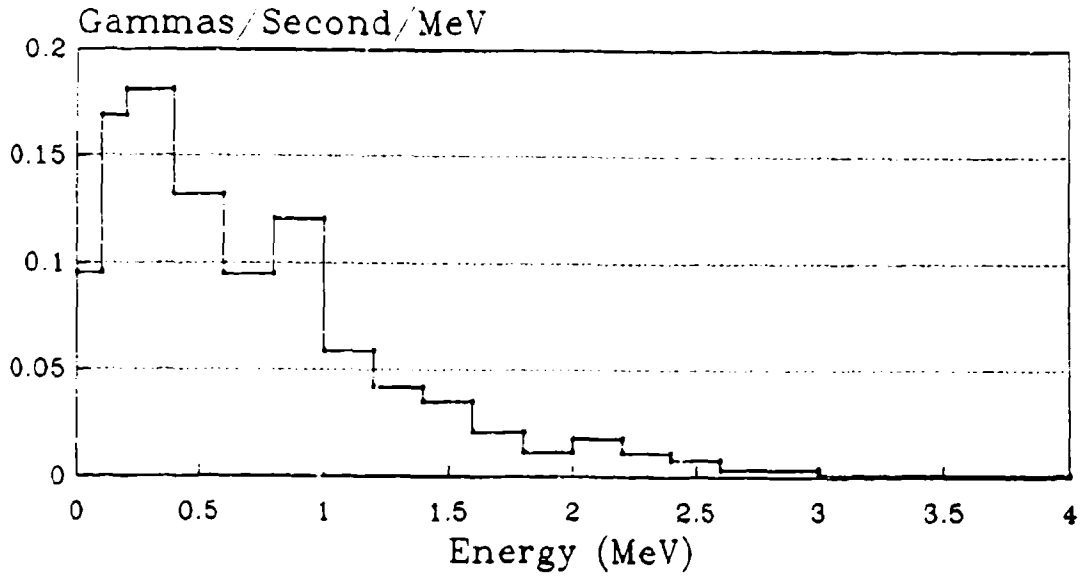
Time = 10 Minutes
U-238



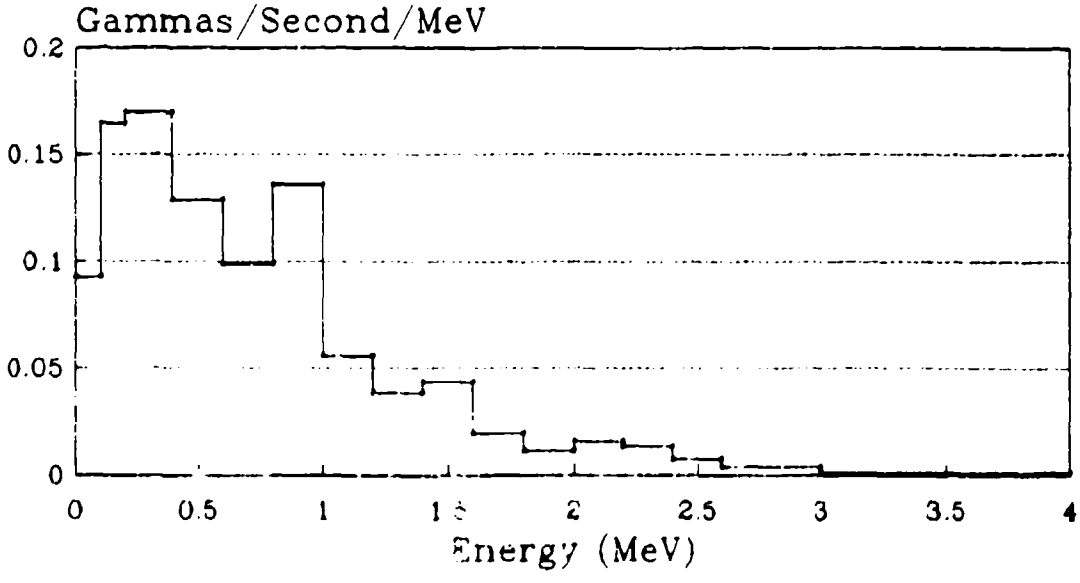
Time = 20 Minutes
U-238



Time = 30 Minutes
U-238

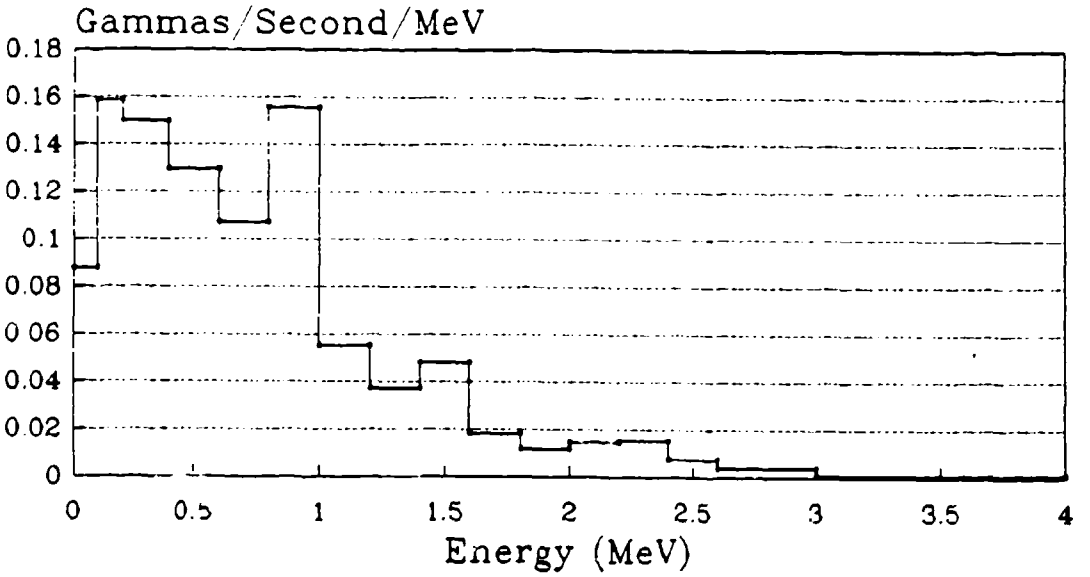


Time = 45 Minutes
U-238



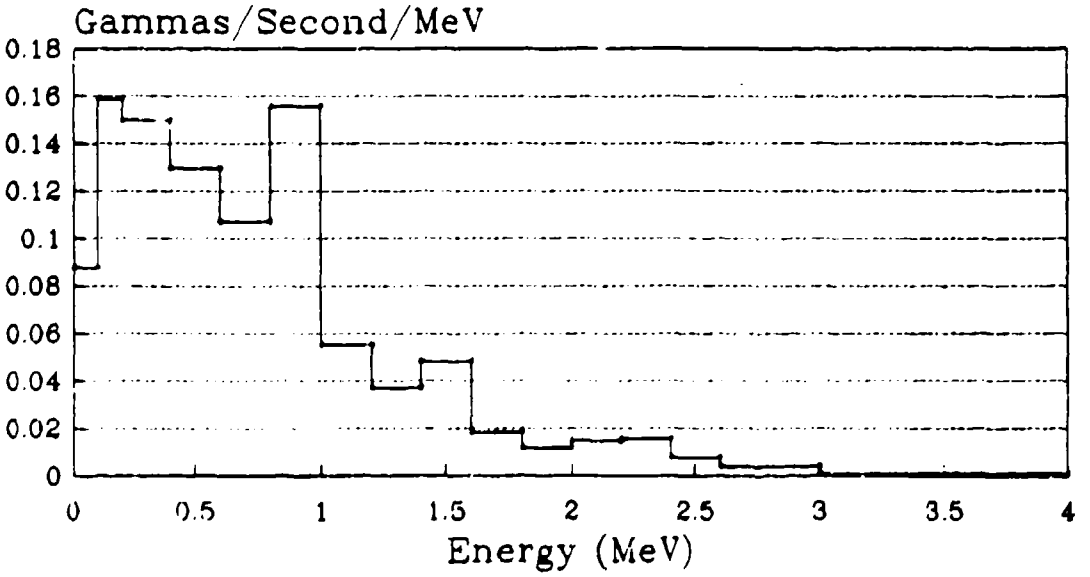
Time = 1 Hour

U-238

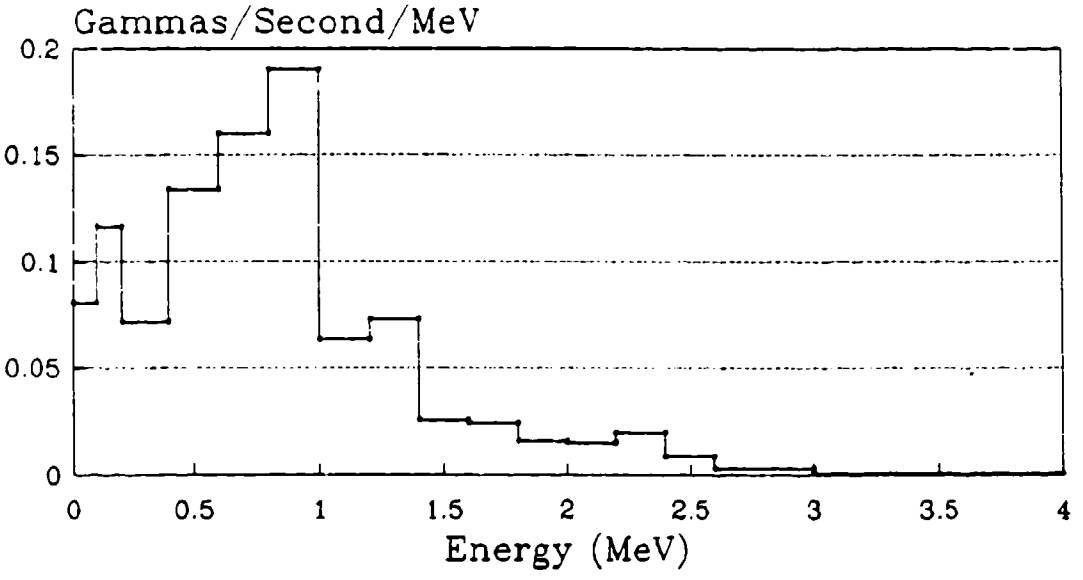


Time = 2 Hours

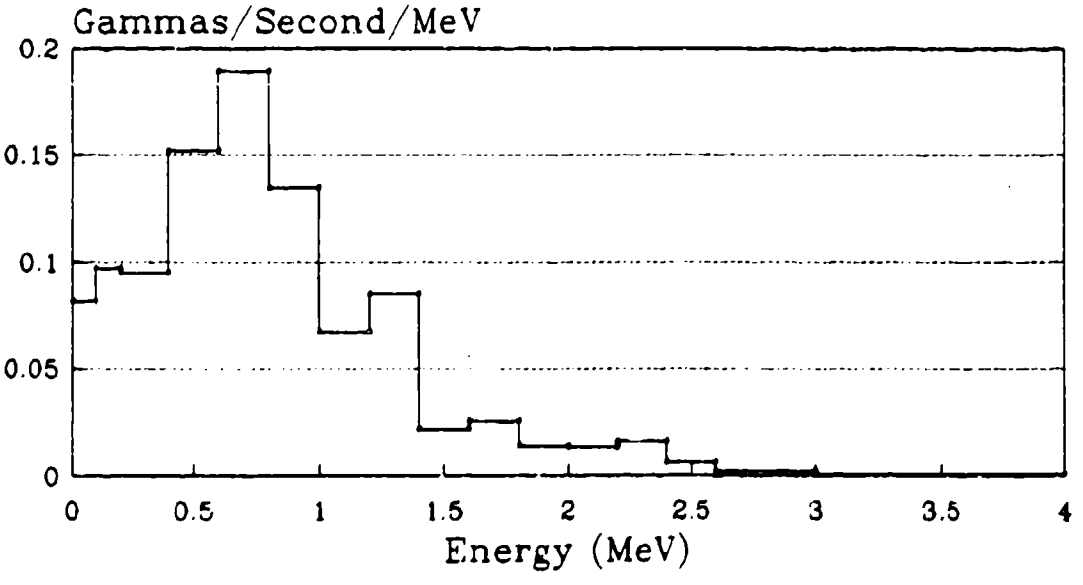
U-238



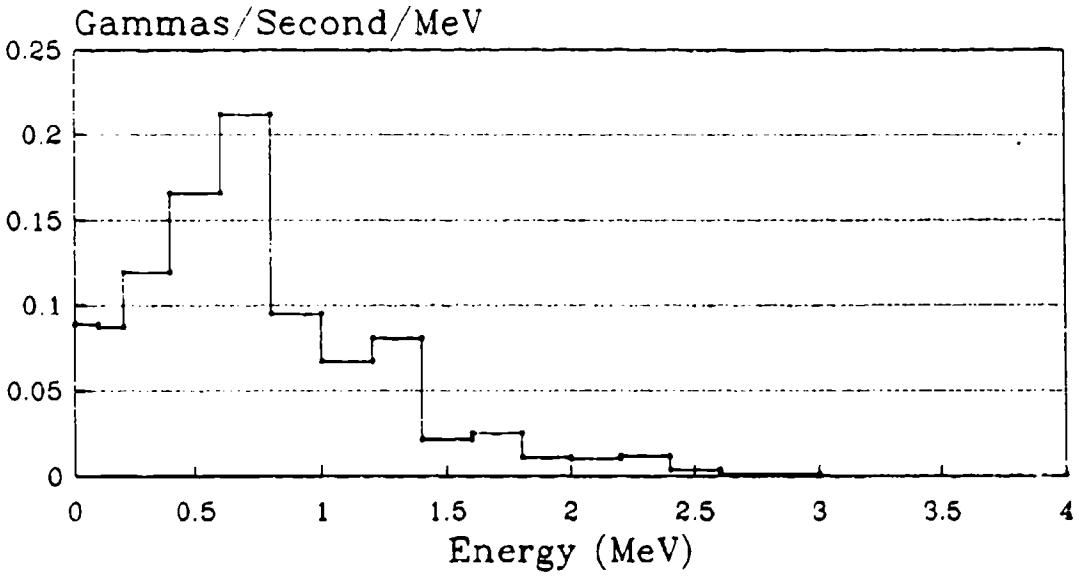
Time = 4 Hours
U-238



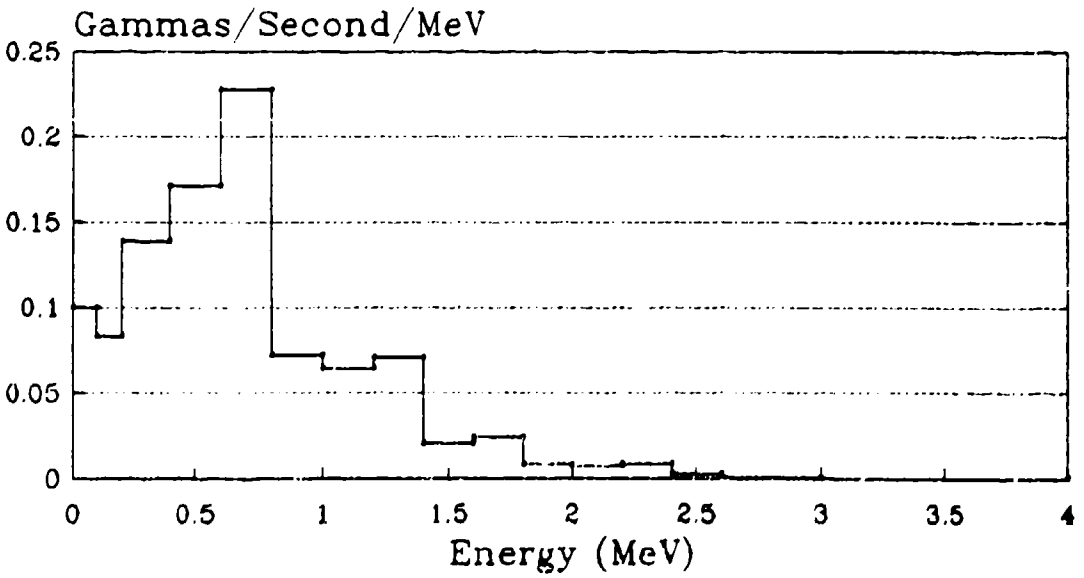
Time = 6 Hours
U-238



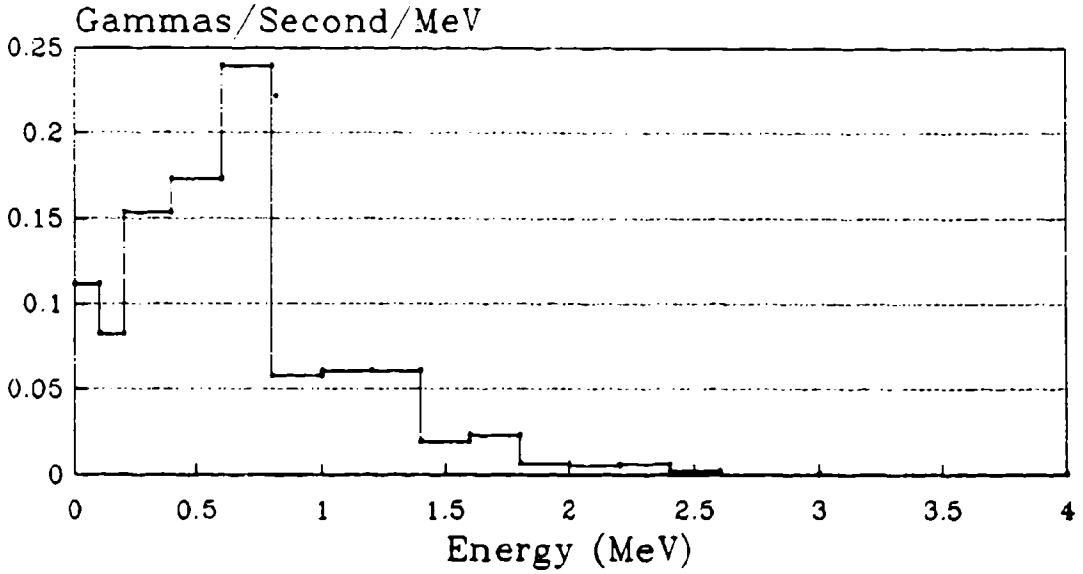
Time = 8 Hours
U-238



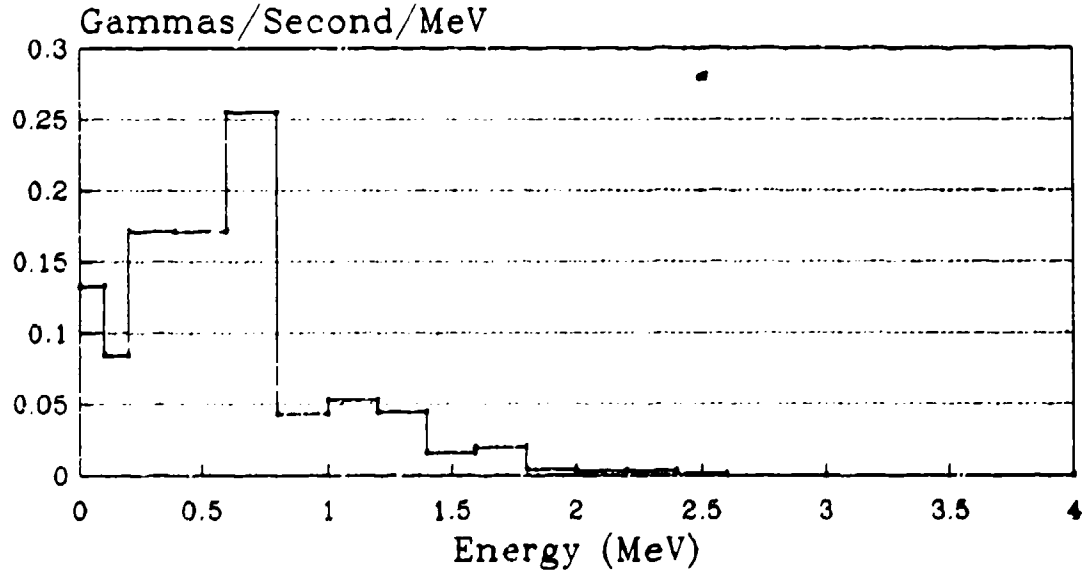
Time = 10 Hours
U-238



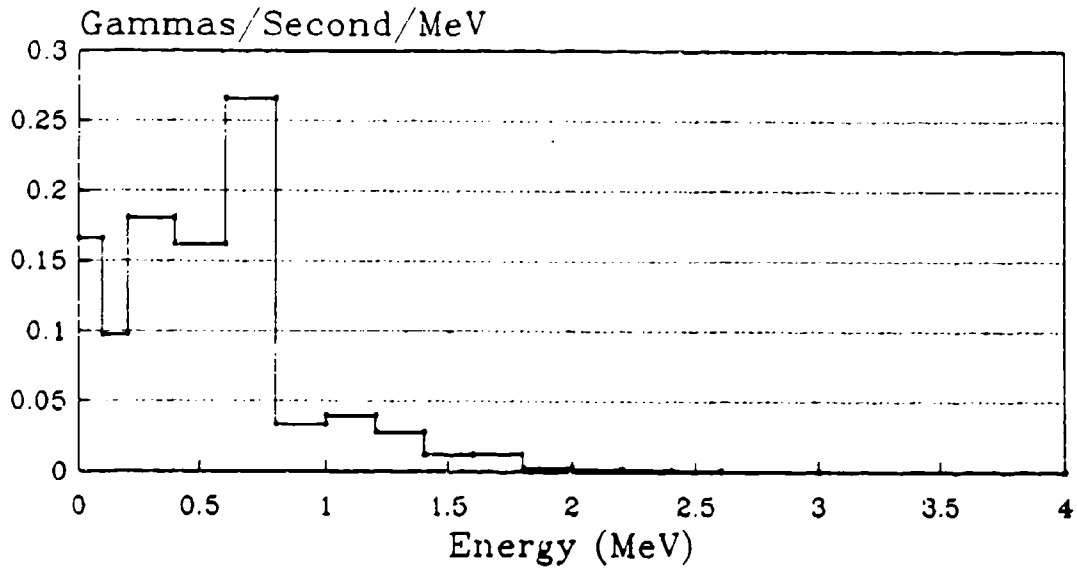
Time = 12 Hours
U-238



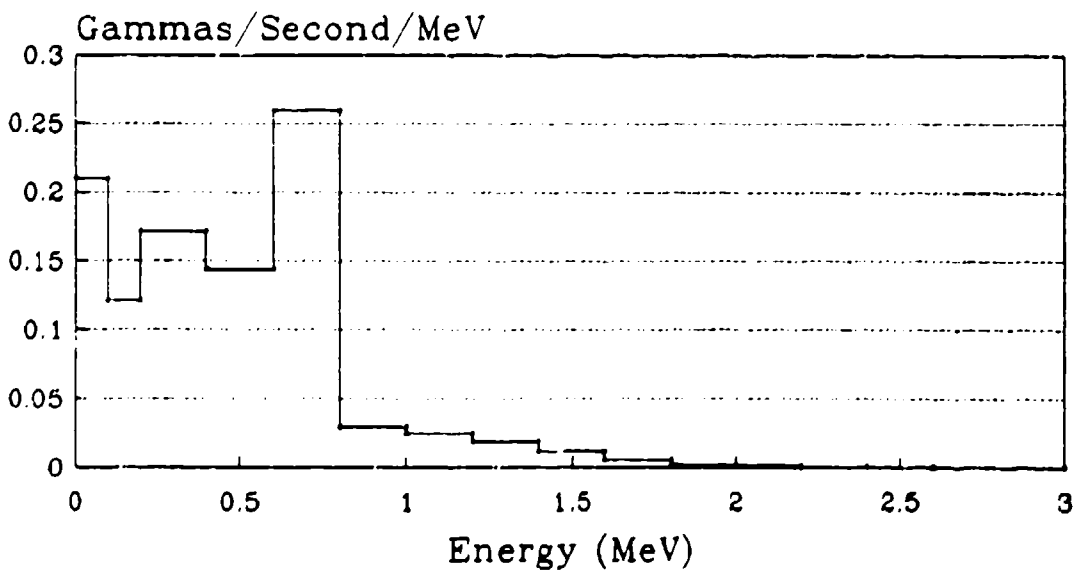
Time = 16 Hours
U-238



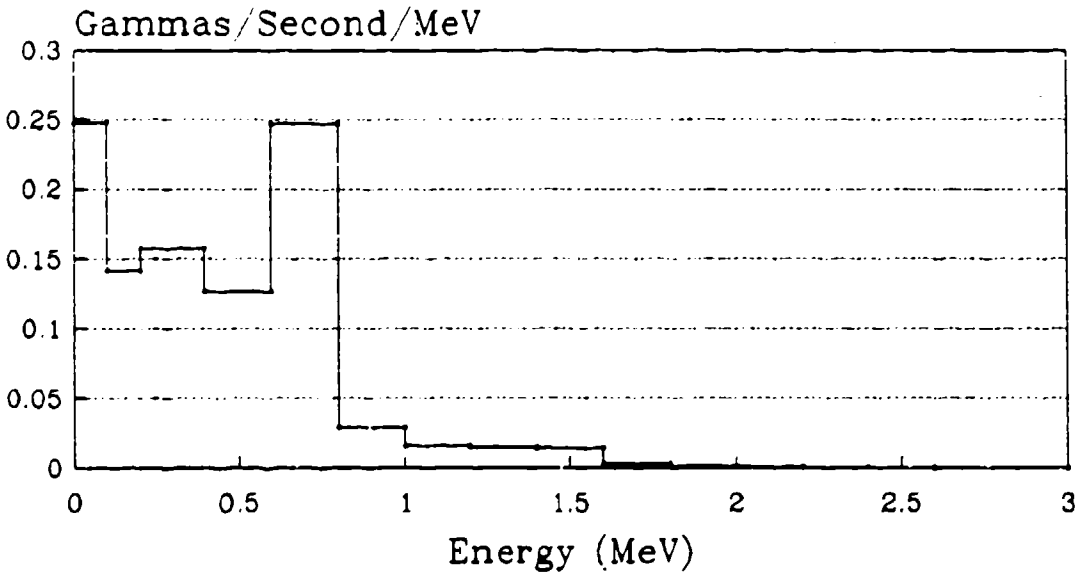
Time = 24 Hours
U-238



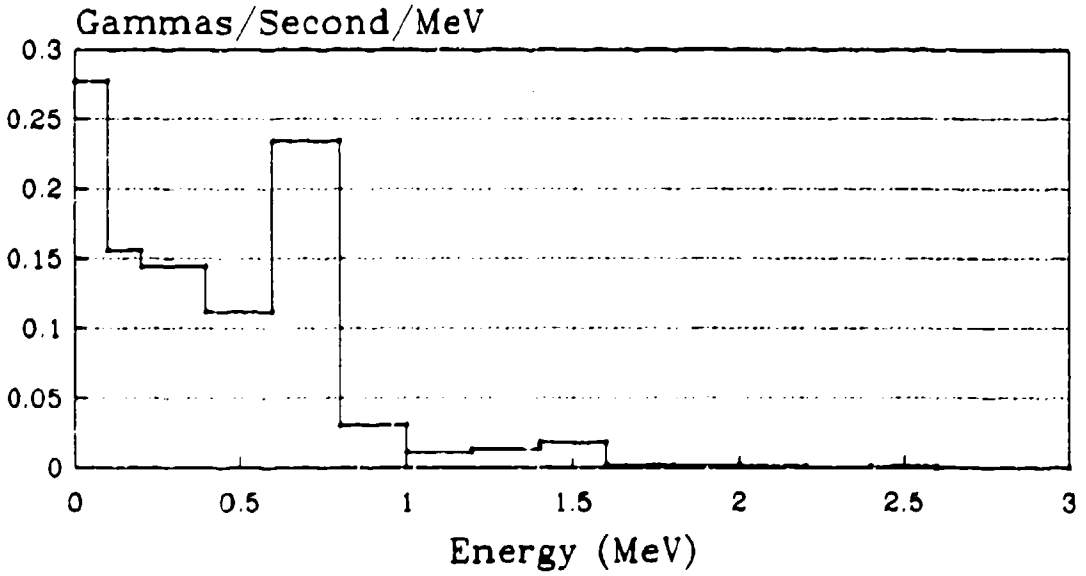
Time = 36 Hours
U-238



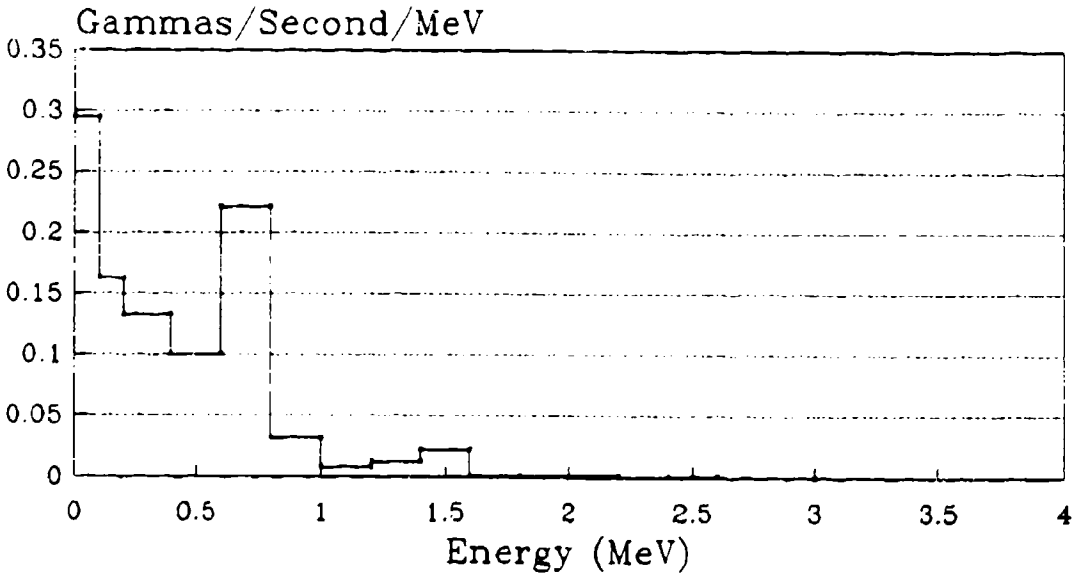
Time = 48 Hours
U-238



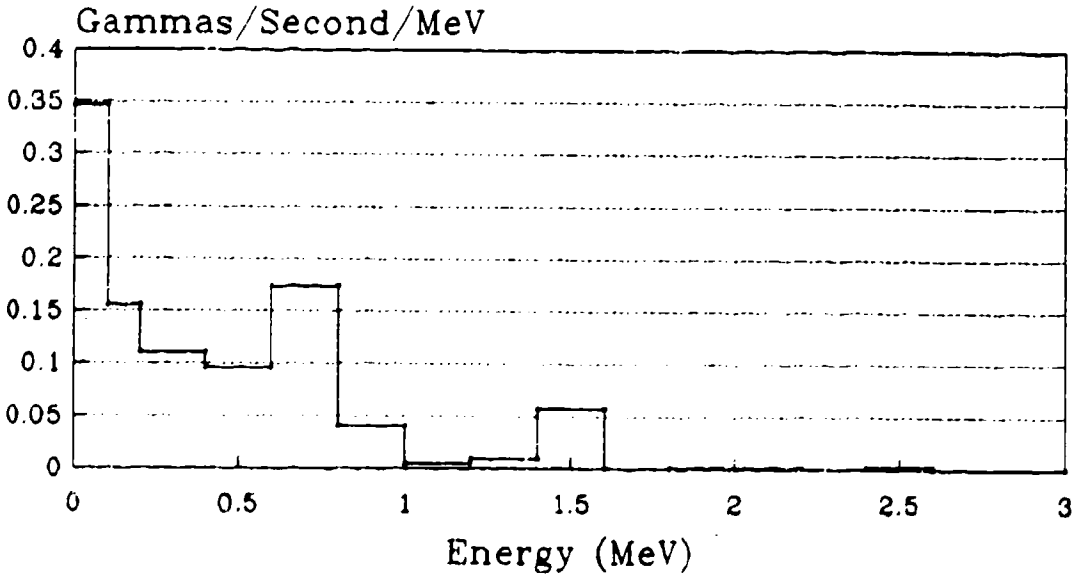
Time = 60 Hours
U-238



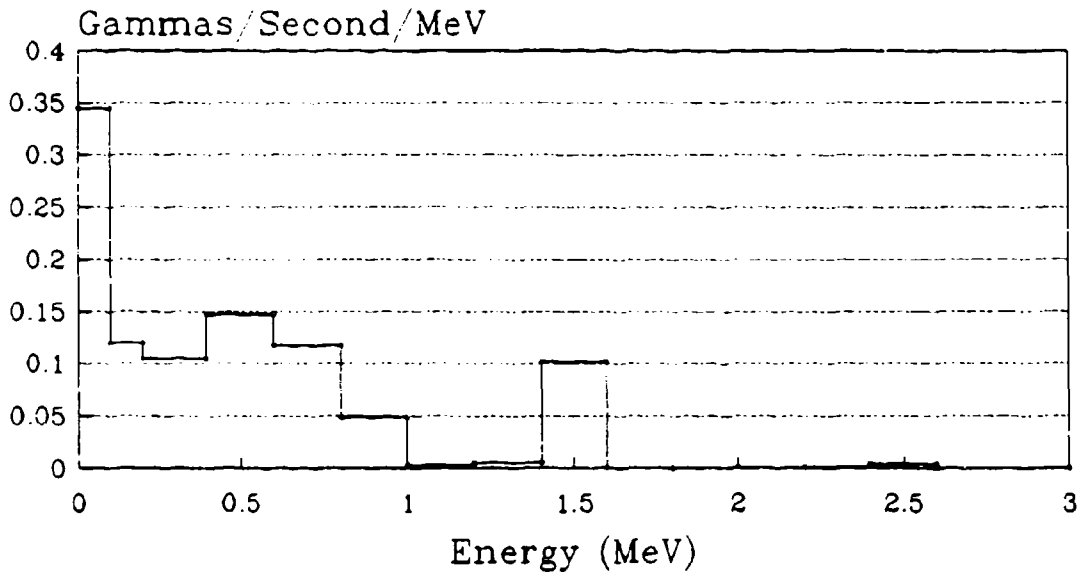
Time = 72 Hours
U-238



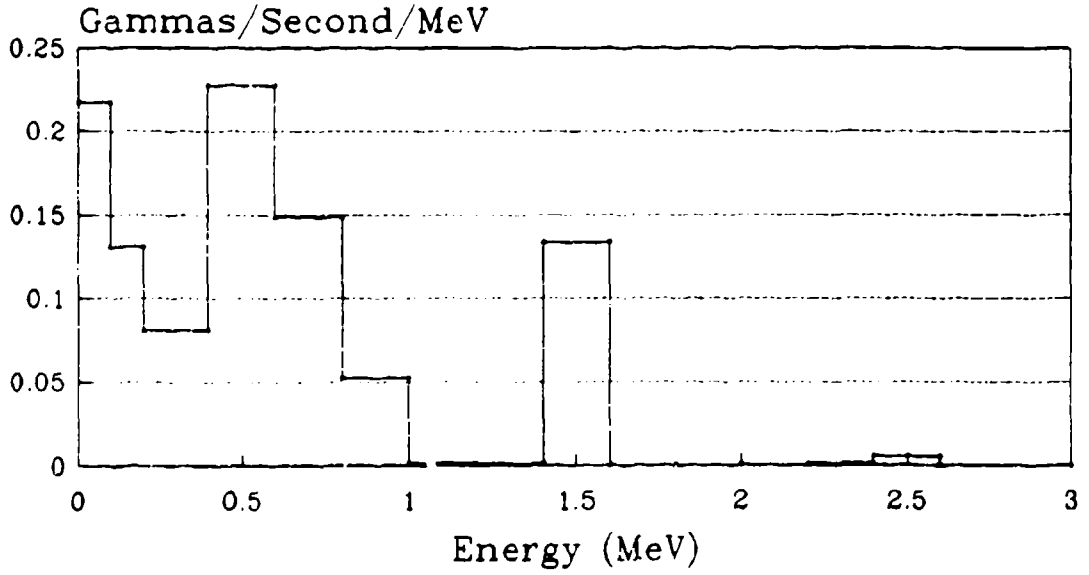
Time = 7 Days
U-238



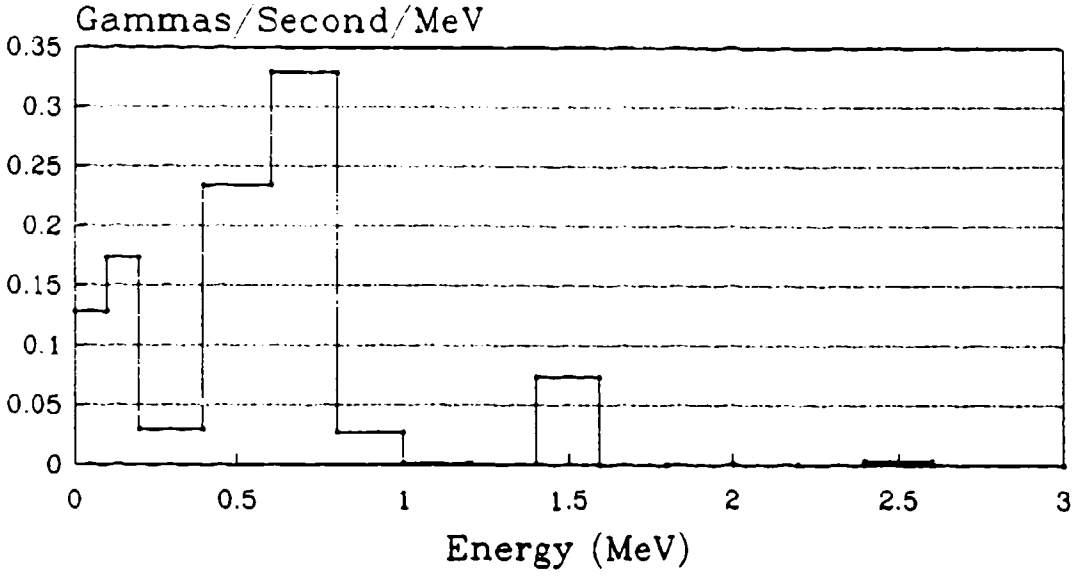
Time = 14 Days
U-238



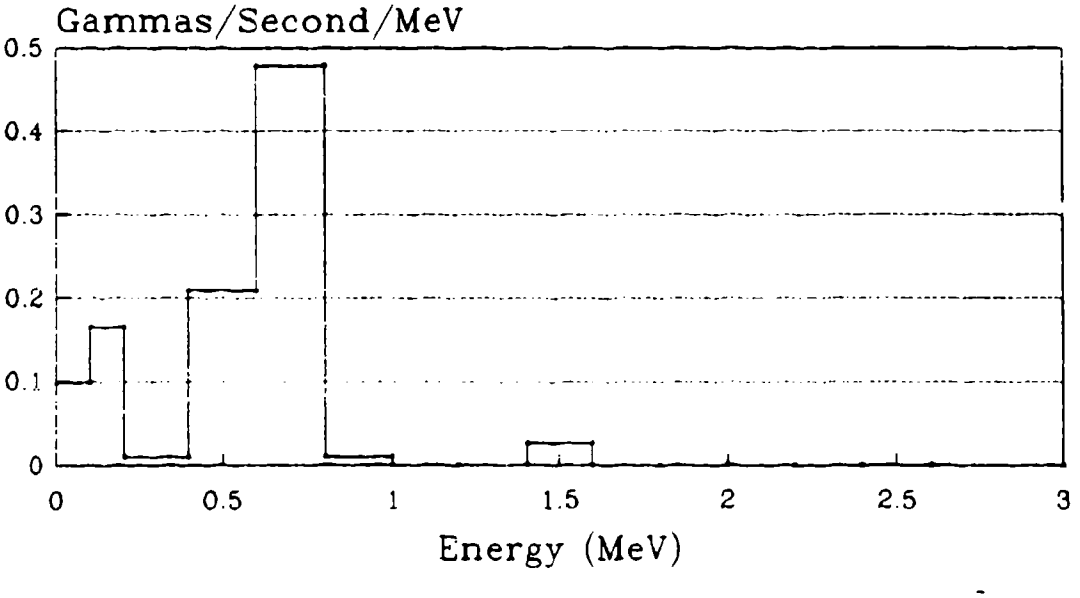
Time = 30 Days
U-238



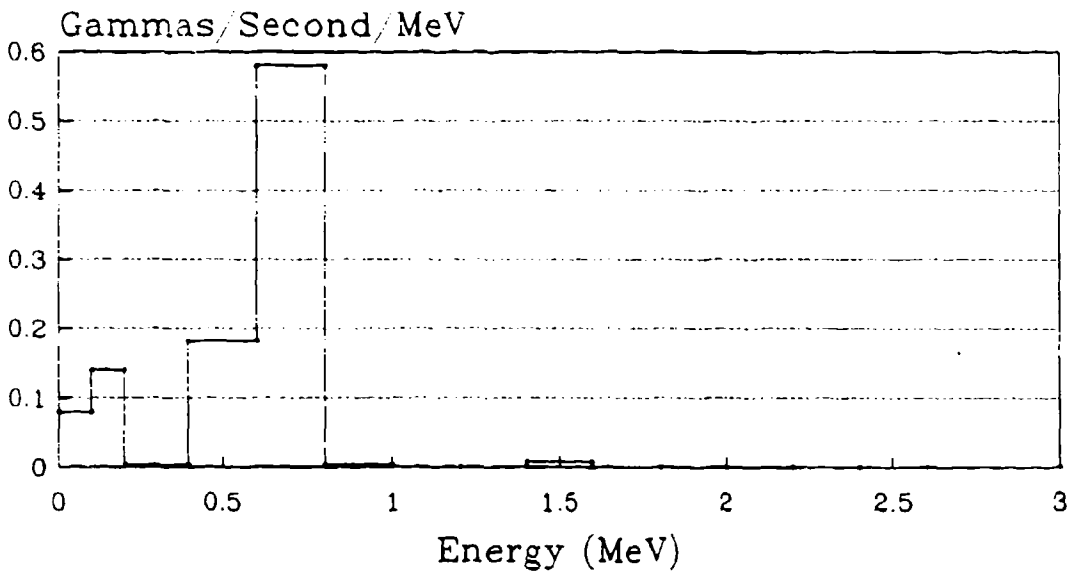
Time = 60 Days
U-238



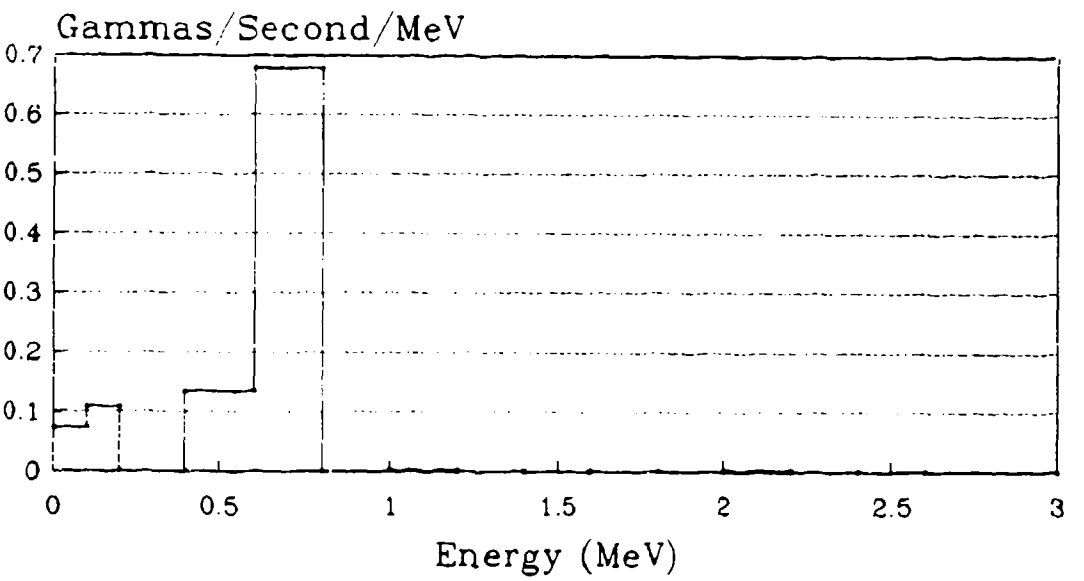
Time = 90 Days
U-238



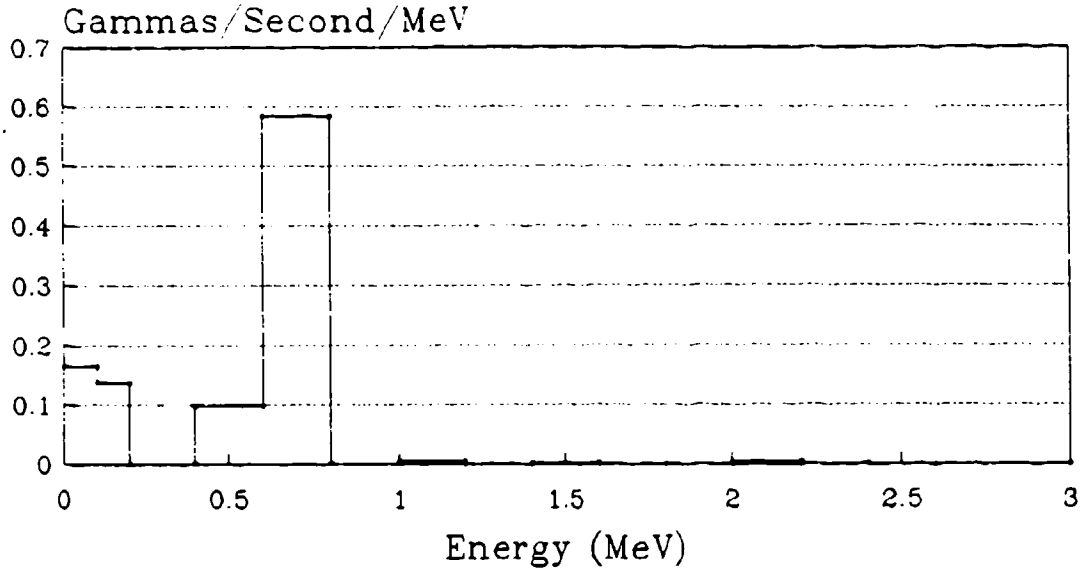
Time = 4 Months
U-238



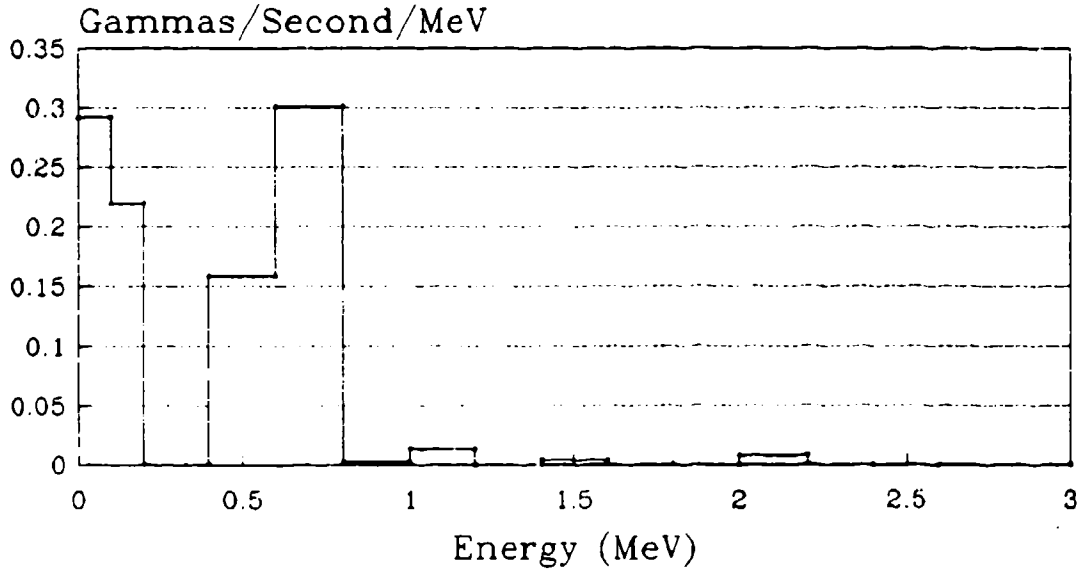
Time = 6 Months
U-238



Time = 1 Year
U-238

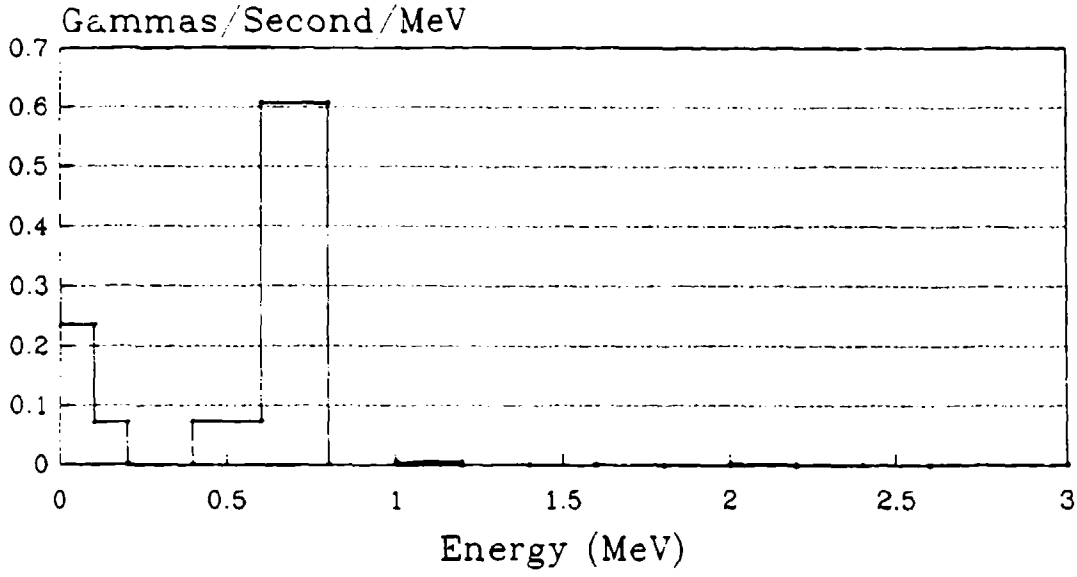


Time = 2 Years
U-238



Time = 5 Years

U-238



Appendix D

MeV Per Second Per Energy Bin

The following tables list the number of MeV emitted per second per energy bin. To determine the approximate number of gamma rays in each bin, divide the MeV/second by the midpoint energy. The data can be used to determine a Source Conversion Coefficient by using some means of summing the scattering photon contribution to exposure rates, i.e. Monte Carlo calculation.

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E (MeV)	10 Minutes	20 Minutes	30 Minutes	45 Minutes	1 Hour
.05	3.437e-06	1.587e-06	9.662e-07	5.611e-07	3.808e-07
.15	1.274e-05	6.864e-06	4.459e-06	2.771e-06	1.937e-06
.3	5.361e-05	3.529e-05	2.453e-05	1.475e-05	9.277e-06
.5	5.839e-05	3.249e-05	2.220e-05	1.427e-05	1.001e-05
.7	7.113e-05	3.632e-05	2.478e-05	1.705e-05	1.289e-05
.9	1.134e-04	6.761e-05	4.733e-05	3.231e-05	2.449e-05
1.1	6.715e-05	3.476e-05	2.260e-05	1.361e-05	9.223e-06
1.3	6.412e-05	3.312e-05	2.172e-05	1.365e-05	9.439e-06
1.5	5.095e-05	3.001e-05	2.301e-05	1.639e-05	1.178e-05
1.7	4.725e-05	2.593e-05	1.619e-05	9.016e-06	5.741e-06
1.9	3.918e-05	2.029e-05	1.265e-05	7.011e-06	4.369e-06
2.1	4.501e-05	2.391e-05	1.556e-05	8.904e-06	5.461e-06
2.3	2.765e-05	1.600e-05	1.074e-05	6.436e-06	4.262e-06
2.5	2.193e-05	1.349e-05	8.833e-06	5.176e-06	3.447e-06
2.8	3.147e-05	1.584e-05	9.764e-06	5.541e-06	3.681e-06
3.5	3.918e-05	1.731e-05	1.064e-05	5.942e-06	3.541e-06
4.5	7.228e-06	4.236e-07	9.559e-08	5.324e-08	3.669e-08
5.5	6.076e-07	2.047e-08	1.600e-09	8.942e-11	6.016e-12
6.75	3.688e-08	2.542e-10	2.016e-12	1.513e-15	3.039e-17

(MeV)	2 Hours	4 Hours	6 Hours	8 Hours	10 Hours
0.05	1.325e-07	3.687e-08	2.377e-08	2.032e-08	1.868e-08
0.15	6.785e-07	1.954e-07	1.042e-07	7.249e-08	5.858e-08
0.30	2.170e-06	6.139e-07	4.757e-07	4.172e-07	3.707e-07
0.50	3.808e-06	1.309e-06	8.346e-07	6.556e-07	5.435e-07
0.70	6.089e-06	2.579e-06	1.676e-06	1.294e-06	1.065e-06
0.90	1.095e-05	3.192e-06	1.227e-06	5.973e-07	3.685e-07
1.10	3.658e-06	1.217e-06	6.487e-07	4.589e-07	3.600e-07
1.30	3.796e-06	1.539e-06	8.697e-07	5.840e-07	4.249e-07
1.50	3.360e-06	6.031e-07	3.258e-07	2.327e-07	1.725e-07
1.70	2.041e-06	7.368e-07	3.966e-07	2.689e-07	2.058e-07
1.90	1.397e-06	4.157e-07	2.143e-07	1.367e-07	9.316e-08
2.10	1.435e-06	3.403e-07	1.447e-07	8.649e-08	5.810e-08
2.30	1.560e-06	4.758e-07	2.199e-07	1.313e-07	8.594e-08
2.50	1.462e-06	4.993e-07	1.910e-07	8.526e-08	4.570e-08
2.80	1.355e-06	3.466e-07	1.268e-07	5.426e-08	2.496e-08
3.50	8.621e-07	2.547e-07	9.513e-08	3.661e-08	1.453e-08
4.50	1.459e-08	5.938e-09	2.882e-09	1.507e-09	8.339e-10
5.50	1.147e-14	2.586e-16	3.904e-17	8.982e-18	2.160e-18
6.75	2.289e-18	7.799e-20	1.850e-20	1.396e-20	1.233e-20

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E (MeV)	12 Hours	16 Hours	24 Hours	36 Hours	48 Hours
0.05	1.765e-08	1.631e-08	1.452e-08	1.250e-08	1.093e-08
0.15	5.132e-08	4.331e-08	3.390e-08	2.523e-08	1.997e-08
0.30	3.312e-07	2.684e-07	1.856e-07	1.186e-07	8.323e-08
0.50	4.610e-07	3.479e-07	2.284e-07	1.467e-07	1.032e-07
0.70	9.059e-07	7.011e-07	4.909e-07	3.397e-07	2.567e-07
0.90	2.735e-07	1.960e-07	1.284e-07	7.894e-08	5.597e-08
1.10	2.917e-07	1.978e-07	9.941e-08	4.492e-08	2.600e-08
1.30	3.201e-07	1.908e-07	8.035e-08	3.697e-08	2.640e-08
1.50	1.298e-07	7.761e-08	3.809e-08	2.810e-08	2.897e-08
1.70	1.657e-07	1.127e-07	5.447e-08	1.983e-08	8.294e-09
1.90	6.573e-08	3.594e-08	1.549e-08	7.618e-09	4.769e-09
2.10	4.025e-08	2.010e-08	6.572e-09	3.168e-09	2.637e-09
2.30	5.746e-08	2.608e-08	6.427e-09	2.178e-09	1.654e-09
2.50	2.864e-08	1.466e-08	5.734e-09	2.808e-09	2.392e-09
2.70	1.228e-08	3.983e-09	1.344e-09	5.803e-10	2.825e-10
3.50	5.973e-09	1.130e-09	7.154e-11	1.949e-11	1.949e-11
4.50	4.802e-10	1.702e-10	2.341e-11	1.237e-12	6.554e-14
5.50	5.688e-19	1.072e-19	7.211e-20	6.047e-20	5.072e-20
6.75	1.126e-20	9.935e-21	8.557e-21	7.257e-21	6.181e-21

E (MeV)	60 Hours	72 Hours	7 Days	14 Days	30 Days
0.05	9.697e-09	8.687e-09	4.331e-09	1.669e-09	3.6317e-10
0.15	1.656e-08	1.416e-08	5.639e-09	1.646e-09	5.8972e-10
0.30	6.259e-08	4.962e-08	1.784e-08	6.979e-09	1.7200e-09
0.50	7.615e-08	5.856e-08	2.314e-08	1.518e-08	7.2801e-09
0.70	2.053e-07	1.707e-07	6.249e-08	1.694e-08	6.5334e-09
0.90	4.431e-08	3.762e-08	1.830e-08	8.263e-09	2.9009e-09
1.10	1.756e-08	1.292e-08	3.270e-09	9.035e-10	2.0038e-10
1.30	2.182e-08	1.868e-08	6.405e-09	1.388e-09	1.2324e-10
1.50	3.098e-08	3.283e-08	3.827e-08	3.063e-08	1.1963e-08
1.70	4.132e-09	2.439e-09	4.063e-10	8.482e-11	4.7714e-12
1.90	3.365e-09	2.593e-09	9.037e-10	2.396e-10	4.3954e-11
2.10	2.395e-09	2.196e-09	1.117e-09	3.719e-10	7.1517e-11
2.30	1.453e-09	1.305e-09	7.340e-10	4.367e-10	1.7222e-10
2.50	2.364e-09	2.388e-09	2.368e-09	1.817e-09	7.5082e-10
2.80	1.558e-10	1.012e-10	5.045e-11	3.464e-11	1.4764e-11
3.50	2.008e-11	2.051e-11	2.031e-11	1.523e-11	6.5363e-12
4.50	3.474e-15	1.843e-16	6.356e-20	5.192e-21	1.6937e-23
5.50	4.252e-20	3.561e-20	8.473e-21	6.567e-22	1.7806e-24
6.75	5.244e-21	4.432e-21	1.051e-21	7.165e-23	1.3207e-25

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E (MeV)	60 Days	90 Days	120 Days	180 Days	1 Year
0.05	6.971e-11	2.943e-11	1.686e-11	7.705e-12	3.065e-12
0.15	2.975e-10	1.567e-10	8.506e-11	2.939e-11	7.124e-12
0.30	2.112e-10	3.641e-11	8.098e-12	1.305e-12	1.872e-13
0.50	2.864e-09	1.507e-09	8.671e-10	3.245e-10	6.385e-11
0.70	5.165e-09	4.098e-09	3.191e-09	1.872e-09	3.449e-10
0.90	5.641e-10	1.127e-10	2.445e-11	3.414e-12	1.735e-12
1.10	5.131e-11	2.439e-11	1.760e-11	1.419e-11	9.905e-12
1.30	2.209e-11	8.143e-12	4.226e-12	1.837e-12	3.310e-13
1.50	2.283e-09	4.791e-10	9.714e-11	7.725e-12	3.215e-12
1.70	1.279e-12	1.033e-12	9.368e-13	8.148e-13	5.668e-13
1.90	9.767e-12	2.688e-12	1.032e-12	5.127e-13	3.436e-13
2.10	2.924e-11	1.741e-11	1.185e-11	7.573e-12	4.403e-12
2.30	3.374e-11	7.368e-12	2.081e-12	7.265e-13	4.727e-13
2.50	1.421e-10	2.895e-11	6.196e-12	5.875e-13	2.447e-13
2.80	3.121e-12	8.007e-13	3.304e-13	1.983e-13	1.367e-13
3.50	1.324e-12	3.049e-13	1.036e-13	5.156e-14	3.511e-14
4.50	3.686e-28	8.024e-33	0.000e+00	0.000e+00	0.000e+00
5.50	2.652e-29	3.942e-34	0.000e+00	0.000e+00	0.000e+00
6.75	9.421e-31	0.000e+00	0.000e+00	0.000e+00	0.000e+00

E (MeV)	2 Years	5 Years
0.05	1.394e-12	3.644e-13
0.15	2.832e-12	2.908e-13
0.30	6.323e-14	2.394e-14
0.50	2.655e-11	3.493e-12
0.70	5.038e-11	2.741e-11
0.90	8.737e-13	1.133e-13
1.10	4.955e-12	6.322e-13
1.30	8.722e-14	1.095e-14
1.50	1.429e-12	1.385e-13
1.70	2.847e-13	3.621e-14
1.90	1.727e-13	2.197e-14
2.10	1.987e-12	1.926e-13
2.30	2.377e-13	3.026e-14
2.50	1.231e-13	1.567e-14
2.80	6.868e-14	8.710e-15
3.50	1.765e-14	2.246e-15
4.50	0.000e+00	0.000e+00
5.50	0.000e+00	0.000e+00
6.75	0.000e+00	0.000e+00

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E (MeV)	10 Minutes	20 Minutes	30 Minutes	45 Minutes	1 Hour
0.05	2.520e-06	1.355e-06	8.586e-07	5.192e-07	3.616e-07
0.15	1.140e-05	6.688e-06	4.554e-06	2.894e-06	2.004e-06
0.3	4.686e-05	2.845e-05	1.893e-05	1.135e-05	7.480e-06
0.5	5.598e-05	3.196e-05	2.087e-05	1.345e-05	9.994e-06
0.7	7.872e-05	3.684e-05	2.449e-05	1.634e-05	1.239e-05
0.9	1.340e-04	7.467e-05	4.909e-05	3.308e-05	2.593e-05
1.1	7.195e-05	3.844e-05	2.418e-05	1.423e-05	9.920e-06
1.3	6.792e-05	3.905e-05	2.498e-05	1.521e-05	1.100e-05
1.5	4.671e-05	2.658e-05	1.977e-05	1.503e-05	1.183e-05
1.7	4.674e-05	2.327e-05	1.374e-05	7.966e-06	5.590e-06
1.9	3.954e-05	2.085e-05	1.311e-05	7.791e-06	5.331e-06
2.1	4.362e-05	2.265e-05	1.460e-05	9.255e-06	6.438e-06
2.3	2.998e-05	1.529e-05	1.050e-05	7.518e-06	5.820e-06
2.5	2.581e-05	1.210e-05	9.089e-06	6.416e-06	4.635e-06
2.8	3.788e-05	1.472e-05	8.333e-06	5.430e-06	4.148e-06
3.5	4.150e-05	1.898e-05	9.961e-06	4.659e-06	2.814e-06
4.5	1.642e-05	1.244e-06	2.047e-07	8.552e-08	6.330e-08
5.5	1.924e-06	8.393e-08	5.028e-09	1.477e-10	9.466e-12
6.75	6.583e-08	3.635e-10	2.353e-12	3.495e-15	2.742e-17

E (MeV)	2 Hours	4 Hours	6 Hours	8 Hours	10 Hours
0.05	1.336e-07	4.452e-08	3.065e-08	2.515e-08	2.143e-08
0.15	7.131e-07	2.543e-07	1.489e-07	9.990e-08	7.359e-08
0.3	2.222e-06	4.817e-07	2.933e-07	2.521e-07	2.276e-07
0.5	4.439e-06	1.502e-06	9.056e-07	7.034e-07	5.871e-07
0.7	6.289e-06	2.514e-06	1.516e-06	1.197e-06	1.051e-06
0.9	1.323e-05	4.063e-06	1.520e-06	7.467e-07	4.707e-07
1.1	4.637e-06	1.689e-06	8.569e-07	5.756e-07	4.472e-07
1.3	5.712e-06	2.513e-06	1.422e-06	9.273e-07	6.446e-07
1.5	4.780e-06	1.564e-06	7.104e-07	3.903e-07	2.537e-07
1.7	2.479e-06	1.080e-06	6.253e-07	4.028e-07	2.826e-07
1.9	2.056e-06	7.918e-07	4.286e-07	2.580e-07	1.697e-07
2.1	2.158e-06	7.152e-07	3.554e-07	1.920e-07	1.106e-07
2.3	2.585e-06	9.964e-07	5.416e-07	3.136e-07	1.850e-07
2.5	1.745e-06	7.233e-07	3.892e-07	2.159e-07	1.227e-07
2.8	1.679e-06	3.810e-07	1.373e-07	6.904e-08	4.076e-08
3.5	1.148e-06	3.453e-07	1.193e-07	4.856e-08	2.296e-08
4.5	2.694e-08	1.084e-08	6.139e-09	3.624e-09	2.152e-09
5.5	3.466e-15	1.460e-16	2.858e-17	6.678e-18	1.587e-18
6.75	6.777e-19	2.885e-20	6.909e-21	3.528e-21	2.772e-21

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E (MeV)	12 Hours	16 Hours	24 Hours	36 Hours	48 Hours
0.05	1.865e-08	1.492e-08	1.150e-08	9.969e-09	9.368e-09
0.15	5.872e-08	4.417e-08	3.287e-08	2.455e-08	1.949e-08
0.3	2.072e-07	1.736e-07	1.264e-07	8.515e-08	6.204e-08
0.5	5.027e-07	3.848e-07	2.539e-07	1.595e-07	1.086e-07
0.7	9.532e-07	8.032e-07	5.856e-07	3.846e-07	2.703e-07
0.9	3.460e-07	2.247e-07	1.173e-07	6.134e-08	4.386e-08
1.1	3.676e-07	2.611e-07	1.397e-07	6.104e-08	3.052e-08
1.3	4.618e-07	2.504e-07	9.278e-08	3.968e-08	2.619e-08
1.5	1.844e-07	1.130e-07	5.066e-08	2.501e-08	2.572e-08
1.7	2.121e-07	1.357e-07	6.697e-08	2.531e-08	1.003e-08
1.9	1.205e-07	7.115e-08	3.225e-08	1.182e-08	5.225e-09
2.1	6.765e-08	2.950e-08	8.454e-09	2.945e-09	2.092e-09
2.3	1.109e-07	4.230e-08	8.502e-09	2.083e-09	1.284e-09
2.5	7.165e-08	2.688e-08	6.308e-09	3.006e-09	2.786e-09
2.8	2.564e-08	1.072e-08	1.964e-09	1.946e-10	6.151e-11
3.5	1.204e-08	3.827e-09	4.530e-10	3.367e-11	1.866e-11
4.5	1.281e-09	4.570e-10	5.977e-11	3.033e-12	1.663e-13
5.5	3.902e-19	4.208e-20	1.994e-20	1.712e-20	1.458e-20
6.75	2.584e-21	2.475e-21	2.286e-21	1.958e-21	1.645e-21

E (MeV)	60 Hours	72 Hours	7 Days	14 Days	30 Days
0.05	8.838e-09	8.265e-09	4.218e-09	1.606e-09	3.509e-10
0.15	1.612e-08	1.373e-08	5.440e-09	1.704e-09	6.762e-10
0.3	4.802e-08	3.892e-08	1.467e-08	5.743e-09	1.513e-09
0.5	7.745e-08	5.755e-08	2.008e-08	1.309e-08	5.791e-09
0.7	2.026e-07	1.603e-07	5.374e-08	1.594e-08	7.710e-09
0.9	3.660e-08	3.232e-08	1.679e-08	8.752e-09	3.211e-09
1.1	1.732e-08	1.101e-08	2.240e-09	5.000e-10	5.568e-11
1.3	1.979e-08	1.578e-08	4.989e-09	1.061e-09	6.590e-11
1.5	3.207e-08	3.816e-08	4.624e-08	3.162e-08	1.325e-08
1.7	4.329e-09	2.148e-09	3.284e-10	6.521e-11	2.448e-12
1.9	2.955e-09	2.079e-09	6.796e-10	1.444e-10	1.636e-11
2.1	1.830e-09	1.656e-09	7.852e-10	2.242e-10	3.203e-11
2.3	1.116e-09	1.047e-09	7.456e-10	4.608e-10	1.830e-10
2.5	2.778e-09	2.779e-09	2.602e-09	1.981e-09	8.656e-10
2.8	5.309e-11	5.384e-11	5.642e-11	4.399e-11	1.765e-11
3.5	1.951e-11	2.066e-11	2.288e-11	1.745e-11	7.369e-12
4.5	9.702e-15	5.912e-16	1.772e-20	1.448e-21	4.730e-24
5.5	1.231e-20	1.033e-20	2.356e-21	3.688e-22	7.490e-23
6.75	1.372e-21	1.142e-21	2.610e-22	1.975e-23	5.412e-26

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E (MeV)	60 Days	90 Days	120 Days	180 Days	1 Year
0.05	6.748e-11	2.955e-11	1.616e-11	7.749e-12	3.883e-12
0.15	3.379e-10	1.763e-10	9.580e-11	3.483e-11	1.019e-11
0.30	2.055e-10	3.640e-11	7.918e-12	1.140e-12	1.390e-13
0.50	1.681e-09	7.143e-10	3.756e-10	1.257e-10	9.244e-12
0.70	6.407e-09	5.206e-09	4.118e-09	2.451e-09	4.260e-10
0.90	5.770e-10	1.150e-10	2.480e-11	1.512e-12	1.833e-13
1.10	9.100e-12	3.212e-12	1.965e-12	1.372e-12	9.215e-13
1.30	1.590e-11	1.083e-11	7.528e-12	3.662e-12	4.427e-13
1.50	2.744e-09	5.418e-10	1.042e-10	7.548e-12	2.827e-12
1.70	1.275e-13	9.930e-14	8.935e-14	7.690e-14	5.321e-14
1.90	3.117e-12	7.124e-13	1.900e-13	4.818e-14	2.913e-14
2.10	2.092e-11	1.942e-11	1.805e-11	1.559e-11	9.930e-12
2.30	3.541e-11	6.957e-12	1.422e-12	1.174e-13	4.427e-14
2.50	1.671e-10	3.241e-11	6.342e-12	2.734e-13	2.316e-14
2.80	3.210e-12	6.549e-13	1.511e-13	2.757e-14	1.559e-14
3.50	1.424e-12	2.821e-13	6.026e-14	7.086e-15	3.279e-15
4.50	1.032e-28	2.252e-33	0.000e+00	0.000e+00	0.000e+00
5.50	5.606e-24	4.197e-25	3.142e-26	1.761e-28	0.000e+00
6.75	8.494e-31	0.000e+00	0.000e+00	0.000e+0	0.000e+00

(MeV)	2 Years	5 Years
0.05	1.5876e-12	2.3010e-13
0.15	3.9263e-12	2.8181e-13
0.30	1.5999e-14	5.9830e-15
0.50	2.6434e-12	4.2606e-13
0.70	3.6968e-11	2.3428e-11
0.90	8.8609e-14	1.0871e-14
1.10	4.6510e-13	5.9847e-14
1.30	3.3898e-14	2.2321e-15
1.50	1.1395e-12	8.2932e-14
1.70	2.6771e-14	3.4197e-15
1.90	1.4522e-14	1.7987e-15
2.10	4.0753e-12	2.8148e-13
2.30	2.2295e-14	2.8506e-15
2.50	1.1566e-14	1.4772e-15
2.80	7.5843e-15	8.9940e-16
3.50	1.6520e-15	2.1135e-16
4.50	0.0000e+00	0.0000e+00
5.50	0.0000e+00	0.0000e+00
6.75	0.0000e+00	0.0000e+00

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E (MeV)	10 Minutes	20 Minutes	30 Minutes	45 Minutes	1 Hour
0.05	3.873e-06	1.563e-06	1.005e-06	6.199e-07	4.136e-07
0.15	1.724e-05	8.371e-06	5.342e-06	3.308e-06	2.239e-06
0.30	5.509e-05	3.320e-05	2.294e-05	1.367e-05	8.454e-06
0.50	7.799e-05	4.280e-05	2.787e-05	1.726e-05	1.220e-05
0.70	8.692e-05	4.289e-05	2.799e-05	1.852e-05	1.412e-05
0.90	1.293e-04	6.710e-05	4.582e-05	3.276e-05	2.639e-05
1.10	7.861e-05	4.263e-05	2.731e-05	1.643e-05	1.143e-05
1.30	6.592e-05	3.584e-05	2.287e-05	1.336e-05	9.066e-06
1.50	3.538e-05	2.575e-05	2.207e-05	1.752e-05	1.354e-05
1.70	4.188e-05	2.279e-05	1.485e-05	8.826e-06	5.815e-06
1.90	2.173e-05	1.313e-05	8.943e-06	5.753e-06	4.191e-06
2.10	3.833e-05	2.382e-05	1.560e-05	8.977e-06	5.739e-06
2.30	1.886e-05	1.283e-05	1.062e-05	8.275e-06	6.578e-06
2.50	1.838e-05	1.172e-05	8.113e-06	5.067e-06	3.499e-06
2.80	3.343e-05	1.199e-05	8.015e-06	5.602e-06	4.105e-06
3.50	2.876e-05	1.249e-05	7.693e-06	4.551e-06	2.966e-06
4.50	5.664e-06	5.979e-07	1.344e-07	5.835e-08	4.308e-08
5.50	7.335e-07	3.851e-08	2.731e-09	9.387e-11	5.223e-12
6.75	1.125e-07	7.391e-10	5.116e-12	2.999e-15	2.147e-17

E (MeV)	2 Hours	4 hours	6 hours	8 hours	10 hours
0.05	1.431e-07	4.908e-08	2.784e-08	2.155e-08	1.906e-08
0.15	7.447e-07	2.131e-07	9.932e-08	6.320e-08	4.751e-08
0.30	1.899e-06	5.248e-07	3.883e-07	3.467e-07	3.172e-07
0.50	4.814e-06	1.634e-06	1.037e-06	8.001e-07	6.523e-07
0.70	6.754e-06	2.741e-06	1.809e-06	1.433e-06	1.212e-06
0.90	1.347e-05	4.185e-06	1.654e-06	8.274e-07	4.926e-07
1.10	4.730e-06	1.705e-06	1.007e-06	7.128e-07	5.377e-07
1.30	4.215e-06	2.311e-06	1.507e-06	1.014e-06	6.999e-07
1.50	4.724e-06	9.297e-07	4.430e-07	3.098e-07	2.342e-07
1.70	2.307e-06	1.001e-06	5.892e-07	4.119e-07	3.128e-07
1.90	1.943e-06	7.252e-07	3.580e-07	1.991e-07	1.185e-07
2.10	2.038e-06	7.541e-07	3.782e-07	2.065e-07	1.179e-07
2.30	3.111e-06	1.102e-06	4.972e-07	2.560e-07	1.454e-07
2.50	1.503e-06	5.310e-07	2.076e-07	9.127e-08	4.684e-08
2.80	1.485e-06	4.087e-07	1.541e-07	6.334e-08	2.790e-08
3.50	1.030e-06	3.556e-07	1.385e-07	5.555e-08	2.304e-08
4.50	1.876e-08	7.297e-09	3.815e-09	2.125e-09	1.220e-09
5.50	8.457e-15	1.107e-16	7.372e-18	1.560e-18	3.725e-19
6.75	1.792e-18	2.607e-20	1.153e-21	2.442e-22	2.773e-22

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E (MeV)	12 Hours	16 hours	24 hours	36 hours	48 hours
0.05	1.765e-08	1.587e-08	1.375e-08	1.187e-08	1.054e-08
0.15	3.893e-08	3.026e-08	2.415e-08	2.061e-08	1.809e-08
0.30	2.912e-07	2.465e-07	1.793e-07	1.164e-07	8.037e-08
0.50	5.477e-07	4.108e-07	2.677e-07	1.633e-07	1.079e-07
0.70	1.059e-06	8.563e-07	6.163e-07	4.117e-07	2.944e-07
0.90	3.298e-07	1.863e-07	9.893e-08	5.984e-08	4.447e-08
1.10	4.209e-07	2.779e-07	1.428e-07	6.138e-08	2.935e-08
1.30	4.970e-07	2.751e-07	1.187e-07	5.521e-08	3.320e-08
1.50	1.812e-07	1.144e-07	5.891e-08	3.898e-08	3.606e-08
1.70	2.459e-07	1.573e-07	6.766e-08	2.165e-08	8.477e-09
1.90	7.500e-08	3.630e-08	1.517e-08	6.988e-09	4.060e-09
2.10	7.060e-08	3.012e-08	1.088e-08	5.334e-09	3.657e-09
2.30	8.816e-08	3.598e-08	7.432e-09	1.726e-09	1.234e-09
2.50	2.810e-08	1.389e-08	5.588e-09	2.797e-09	2.404e-09
2.80	1.320e-08	3.579e-09	4.472e-10	9.816e-11	7.999e-11
3.50	9.947e-09	2.112e-09	1.557e-10	2.084e-11	1.881e-11
4.50	7.161e-10	2.564e-10	3.525e-11	1.870e-12	9.977e-14
5.50	9.284e-20	1.031e-20	5.017e-21	4.912e-21	4.561e-21
6.75	4.268e-22	6.536e-22	7.630e-22	6.779e-22	5.673e-22

(MeV)	60 Hours	72 hours	7 days	14 days	30 days
0.05	9.446e-09	8.514e-09	4.260e-09	1.710e-09	3.480e-10
0.15	1.594e-08	1.409e-08	5.742e-09	1.786e-09	6.295e-10
0.30	5.903e-08	4.590e-08	1.628e-08	5.281e-09	1.559e-09
0.50	7.628e-08	5.764e-08	2.358e-08	1.461e-08	7.296e-09
0.70	2.236e-07	1.784e-07	5.977e-08	1.635e-08	6.668e-09
0.90	3.702e-08	3.269e-08	1.790e-08	8.681e-09	3.011e-09
1.10	1.602e-08	1.020e-08	2.639e-09	6.072e-10	8.859e-11
1.30	2.333e-08	1.823e-08	6.122e-09	1.251e-09	1.011e-10
1.50	3.667e-08	3.812e-08	4.212e-08	3.052e-08	1.287e-08
1.70	4.112e-09	2.372e-09	3.719e-10	8.195e-11	3.810e-12
1.90	2.824e-09	2.227e-09	8.448e-10	2.137e-10	3.298e-11
2.10	2.940e-09	2.534e-09	1.063e-09	2.839e-10	5.615e-11
2.30	1.144e-09	1.088e-09	7.649e-10	4.591e-10	1.768e-10
2.50	2.415e-09	2.476e-09	2.526e-09	1.896e-09	8.072e-10
2.80	7.566e-11	7.231e-11	5.438e-11	3.693e-11	1.592e-11
3.50	1.986e-11	2.072e-11	2.169e-11	1.626e-11	6.934e-12
4.50	5.325e-15	2.842e-16	6.201e-21	4.175e-22	8.754e-25
5.50	4.007e-21	3.389e-21	5.033e-22	9.176e-24	6.236e-28
6.75	4.715e-22	3.916e-22	8.853e-23	6.561e-24	1.713e-26

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(MeV)	60 Days	90 days	120 Days	180 Days	1 year
0.05	7.426e-11	3.124e-11	1.607e-11	7.448e-12	3.469e-12
0.15	3.015e-10	1.564e-10	8.601e-11	3.276e-11	8.604e-12
0.30	2.042e-10	3.635e-11	7.960e-12	1.144e-12	1.597e-13
0.50	2.710e-09	1.325e-09	7.425e-10	2.731e-10	4.105e-11
0.70	5.337e-09	4.248e-09	3.311e-09	1.936e-09	3.440e-10
0.90	5.665e-10	1.134e-10	2.455e-11	2.513e-12	1.031e-12
1.10	2.127e-11	1.251e-11	1.004e-11	8.322e-12	5.821e-12
1.30	1.576e-11	8.641e-12	5.773e-12	2.706e-12	3.626e-13
1.50	2.542e-09	5.046e-10	1.032e-10	8.326e-12	2.990e-12
1.70	7.288e-13	5.897e-13	5.350e-13	4.717e-13	3.326e-13
1.90	6.726e-12	1.734e-12	6.246e-13	2.981e-13	1.999e-13
2.10	2.450e-11	1.801e-11	1.575e-11	1.336e-11	8.460e-12
2.30	3.458e-11	7.238e-12	1.795e-12	4.478e-13	2.775e-13
2.50	1.577e-10	3.095e-11	6.220e-12	4.321e-13	1.441e-13
2.80	3.281e-12	7.441e-13	2.463e-13	1.215e-13	8.196e-14
3.50	1.387e-12	2.991e-13	8.450e-14	3.116e-14	2.058e-14
4.50	8.318e-30	7.905e-35	0.000e+00	0.000e+00	0.000e+00
5.50	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00
6.75	2.457e-31	0.000e+00	0.000e+00	0.000e+00	0.000e+00

(MeV)	2 Years	5 Years
0.05	1.417e-12	3.242e-13
0.15	3.197e-12	2.970e-13
0.30	3.106e-14	1.201e-14
0.50	1.541e-11	2.023e-12
0.70	4.089e-11	2.344e-11
0.90	5.141e-13	6.511e-14
1.10	2.927e-12	3.721e-13
1.30	6.275e-14	6.234e-15
1.50	1.301e-12	1.136e-13
1.70	1.672e-13	2.126e-14
1.90	1.005e-13	1.278e-14
2.10	3.447e-12	2.380e-13
2.30	1.396e-13	1.777e-14
2.50	7.247e-14	9.219e-15
2.80	4.073e-14	5.147e-15
3.50	1.035e-14	1.317e-15
4.50	0.000e+00	0.000e+00
5.50	0.000e+00	0.000e+00
6.75	0.000e+00	0.000e+00

Appendix E

Source Normalization Constant as a Function of Time

The following tables should be used to calculate exposure rates more accurately than by applying the Way-Wigner approximation to the Source Normalization Constant (SNC) at one hour. Units on the time dependent SNC are $(R/Hr)/(kT/km^2)$.

Time (Hrs)	U-235	Pu-239	U-238
Ø.1667E+ØØ	Ø.4427E+Ø5	Ø.4197E+Ø5	Ø.427ØE+Ø5
Ø.3333E+ØØ	Ø.23Ø4E+Ø5	Ø.2297E+Ø5	Ø.23Ø8E+Ø5
Ø.5ØØØE+ØØ	Ø.1487E+Ø5	Ø.1545E+Ø5	Ø.1561E+Ø5
Ø.75ØØE+ØØ	Ø.9598E+Ø4	Ø.9739E+Ø4	Ø.1Ø16E+Ø5
Ø.1ØØØE+Ø1	Ø.7Ø56E+Ø4	Ø.6756E+Ø4	Ø.7371E+Ø4
Ø.2ØØØE+Ø1	Ø.3163E+Ø4	Ø.254ØE+Ø4	Ø.31Ø5E+Ø4
Ø.4ØØØE+Ø1	Ø.1110E+Ø4	Ø.8195E+Ø3	Ø.1Ø87E+Ø4
Ø.6ØØØE+Ø1	Ø.5656E+Ø3	Ø.4354E+Ø3	Ø.5867E+Ø3
Ø.8ØØØE+Ø1	Ø.365ØE+Ø3	Ø.2967E+Ø3	Ø.39Ø9E+Ø3
Ø.12ØØE+Ø2	Ø.2148E+Ø3	Ø.1839E+Ø3	Ø.2274E+Ø3
Ø.16ØØE+Ø2	Ø.15Ø9E+Ø3	Ø.132ØE+Ø3	Ø.1575E+Ø3
Ø.24ØØE+Ø2	Ø.9Ø3ØE+Ø2	Ø.82ØØE+Ø2	Ø.96Ø9E+Ø2
Ø.36ØØE+Ø2	Ø.531ØE+Ø2	Ø.5143E+Ø2	Ø.58Ø3E+Ø2
Ø.6ØØØE+Ø2	Ø.2798E+Ø2	Ø.2962E+Ø2	Ø.3Ø33E+Ø2
Ø.72ØØE+Ø2	Ø.2285E+Ø2	Ø.246ØE+Ø2	Ø.2456E+Ø2
Ø.168ØE+Ø3	Ø.1Ø12E+Ø2	Ø.1Ø86E+Ø2	Ø.1Ø77E+Ø2
Ø.336ØE+Ø3	Ø.4764E+Ø1	Ø.4991E+Ø1	Ø.4882E+Ø1
Ø.72ØØE+Ø3	Ø.1933E+Ø1	Ø.1878E+Ø1	Ø.1929E+Ø1
Ø.144ØE+Ø4	Ø.7186E+ØØ	Ø.6884E+ØØ	Ø.7Ø17E+ØØ
Ø.216ØE+Ø4	Ø.4118E+ØØ	Ø.3877E+ØØ	Ø.387ØE+ØØ
Ø.288ØE+Ø4	Ø.2879E+ØØ	Ø.26Ø3E+ØØ	Ø.26Ø4E+ØØ
Ø.432ØE+Ø4	Ø.16Ø5E+ØØ	Ø.1374E+ØØ	Ø.1383E+ØØ
Ø.876ØE+Ø4	Ø.28Ø2E-Ø1	Ø.2659E-Ø1	Ø.2511E-Ø1
Ø.1752E+Ø5	Ø.3Ø53E-Ø2	Ø.4Ø63E-Ø2	Ø.416ØE-Ø2
Ø.438ØE+Ø5	Ø.15Ø5E-Ø2	Ø.1984E-Ø2	Ø.164ØE-Ø2

Appendix F

Coefficients for Build Up Factors

These Build Up Factors (BUF) were determined from Kalansky and Bigelow's works and are fit to Taylors fit for BUF. The BUF are accurate within 1% up to 3 mean free paths.

KNRRGY	C1	C2	A1	A2
Ø.Ø5	.13	.ØØ9	22.17969	-21.17969
Ø.15Ø	.286	.Ø144	12.13655	-11.1365
Ø.3ØØ	.16	.1575	941.9688	94Ø.9688
Ø.5ØØ	.186	.ØØ94	11.ØØ8942	-1Ø.Ø8942
Ø.7ØØ	.159	.ØØ85	1Ø.2Ø49	-9.2Ø49
Ø.9ØØ	.Ø5	.Ø95	-27.23407	28.23407
1.1ØØ	.113	.Ø1Ø1	1Ø.67752	-9.677525
1.3ØØ	.12	.ØØ81	7.813421	-6.813421
1.5ØØ	.112	.ØØ11	7.Ø66833	-6.Ø66833
1.7ØØ	.Ø83	.Ø1Ø8	1Ø.1514	-9.1514
1.9ØØ	.Ø41	.Ø196	34.33551	-33.33551
2.1ØØ	.Ø13	.Ø166	-2Ø4.4141	2Ø5.4141
2.3ØØ	.ØØ75	.ØØ585	426.5469	-425.5469
2.5ØØ	.ØØ6	.ØØ525	875.4375	-874.4375
2.8ØØ	.ØØ6	.ØØ61	-5984	5985.
3.5ØØ	.ØØØ2	.ØØØ54	-1487.75	1488.75
4.5ØØ	.ØØ292	.ØØ311	-2Ø89.5	2Ø9Ø.5
5.5ØØ	.ØØØØØ5	.ØØØ12	-2944.	2945.
6.75	.ØØØ5	.ØØØØ21	463.875	-462.875

Appendix G

Cross Sections

The absorption and total cross-sections were interpolated from data in Rockwell's *Reactor Shielding Handbook*. The scattering cross-section is taken as simply $\mu_s = \mu_t - \mu_a$.

MIDPOINT ENRRGY	MUt	MUa
Ø.Ø5	Ø.Ø2Ø5	Ø.ØØ47
Ø.15	Ø.Ø133	Ø.ØØ24
Ø.3	Ø.Ø1Ø7	Ø.ØØ28
Ø.5	Ø.ØØ87	Ø.ØØ29
Ø.7	Ø.ØØ75	Ø.ØØ29
Ø.9	Ø.ØØ67	Ø.ØØ28
1.1	Ø.ØØ6Ø	Ø.ØØ27
1.3	Ø.ØØ55	Ø.ØØ26
1.5	Ø.ØØ51	Ø.ØØ245
1.7	Ø.ØØ47	Ø.ØØ24
1.9	Ø.ØØ45	Ø.ØØ23
2.1	Ø.ØØ44	Ø.ØØ225
2.3	Ø.ØØ4Ø	Ø.ØØ22
2.5	Ø.ØØ38	Ø.ØØ215
2.8	Ø.ØØ36	Ø.ØØ2Ø5
3.5	Ø.ØØ31	Ø.ØØ195
4.5	Ø.ØØ28	Ø.ØØ185
5.5	Ø.ØØ26	Ø.ØØ175
6.75	Ø.ØØ24	Ø.ØØ165

Appendix H

Average Compton Scatter Energies

Energy	Energy Prime
.Ø5	.Ø4596126
.15	.1227656
.3	.2191Ø36
.5	.3293169
.7	.4272675
.9	.517221
1.1	.6Ø1472
1.3	.6814173
1.5	.7579752
1.7	.831784
1.9	.9Ø33Ø61
2.1	.972889
2.3	1.Ø4Ø8
2.5	1.1Ø7251
2.8	1.2Ø4553
3.5	1.422611
4.5	1.717852
5.5	1.999285
6.75	2.336871

Appendix I

Successive Scatter Code

```
DECLARE SUB EXPOFNCT (X, E1!)
DECLARE SUB FIRSTFLIGHT (GpS!(), GT!, MUt!(), MUa!(), ENERGY!(), ACTIVITY!,
LOCALFRACTION!, MCiACTIVITY!, AREACONVERSION!, UNITLENGTH$, SCC!)
DECLARE SUB FIRSTSCATTER (GpS(), GT, MUt(), MUa(),  MUs(), ENERGY(),
ACTIVITY,
 LOCALFRACTION, EPRIME(), EBRACKET(), ALT(), FIRSTSCAT(), ZCMAX, ALTMAX)
DECLARE SUB SCNDSCATTER (GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), EPRI-
ME(),
 EBRACKET(), ALT(), FIRSTSCAT(), SCNDSCAT(), ZCMAX, ALT2())
DECLARE SUB THIRDSCATTER (GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), EPRI-
ME(),
 EBRACKET(), ALT(), FIRSTSCAT(), SCNDSCAT(), THIRDSCAT(), ZCMAX, ALT2())
DECLARE SUB DOSECALC (GpS(), GT, MUt(), MUa(), ENERGY(), ACTIVITY, ALT(),
 FIRSTSCAT(), SCNDSCAT(), THIRDSCAT(), ZCMAX, AREACONVERSION, MCiACTIV-
ITY,
 UNITLENGTH$, SCATCONT, SCNDSCATCONT, THIRDSCATCONT)
'*****************************************************************
*********
'*** This program will calculate the dose at 1 meter above an infinite
***
'*** homogeneous plane of activity.  The gamma ray spectrum of the
***
'*** activity is read in from another file.  The dose is calculated
***
'*** from first flight photons (i.e. uncollided flux), and first,
***
'*** second and third scattered photons.  The energy loss due to
each  ***
'*** successive scatter is dictated by the Klein-Nishina average
***
'*** energy loss per collision formula.  Isotropic scatter is assumed.
***

'*****************************************************************
*********
DIM ENERGY(2Ø), MEVperSEC(2Ø), MUt(2Ø), MUa(2Ø), MUs(2Ø), GpS(2Ø)
DIM EBRACKET(2Ø), EPRIME(2Ø), ALT(5Ø), FIRSTSCAT(5Ø, 2Ø), SCNDSCAT(5Ø,
2Ø)
DIM THIRDSCAT(5Ø, 2Ø), ALT2(5Ø)
GOSUB SETUP
'******************** OPEN FILE OF GAMMA RAY SPECTRUM DATA
***************
   OPEN "C:\THESIS\U8DATA\MEVPRSEC.5Y" FOR INPUT AS #1
   INPUT #1, TITLE$
   TOTALMEV = Ø
   GT = Ø
    FOR I = 1 TO 19
```

```
      INPUT #1, ENERGY(I), MEVperSEC(I)
      GpS(I) = MEVperSEC(I) / ENERGY(I)
      'PRINT ENERGY(I), MEVperSEC(I), GpS(I)
      TOTALMEV = TOTALMEV + MEVperSEC(I)
      GT = GT + GpS(I)
    NEXT I
   CLOSE
'*************************************************************************
********
'********************** OPEN FILE OF CROSS SECTION DATA
******************
   OPEN "C:\THESIS\HOMEPROG\XSECTION" FOR INPUT AS #1
     INPUT #1, XTITLE$
     FOR I = 1 TO 19
      INPUT #1, ENERGY(I), MUt(I), MUa(I)
      MUs(I) = MUt(I) - MUa(I)
     NEXT I
   CLOSE
'*************************************************************************
********
'******** INPUT ENERGY DEPENDENT KLEIN-NISHINA CROSS SECTION DATA
*******
   OPEN "C:\THESIS\HOMEPROG\EPRIME.EIN" FOR INPUT AS #1
     FOR I = 1 TO 19
      INPUT #1, ENERGYDUMMY, EPRIME(I)
      'PRINT EPRIME(I)
     NEXT I
    CLOSE #1
'*************************************************************************
********
 CALL FIRSTFLIGHT(GpS(), GT, MUt(), MUa(), ENERGY(), ACTIVITY, LOCALFRAC-
TION,
  MCiAIVITY, AREACONVERSION, UNITLENGTH$, SCC)
   DATA Ø, .1, .2, .4, .6, .8, 1.Ø, 1.2, 1.4, 1.6, 1.8
   DATA 2.Ø, 2.2, 2.4, 2.6, 3.Ø, 4.Ø, 5.Ø, 6.Ø, 7.5
    FOR I = 1 TO 2Ø
      READ EBRACKET(I)
    NEXT I
   ALTMAX = 15Ø                              'MAXIMUM ALTITUDE AT WHICH
TO CALCULATE
                                             'THE SCATTER RATE
 CALL FIRSTSCATTER(GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), ACTIVITY,
  LOCALFRACTI, EPRIME(), EBRACKET(), ALT(), FIRSTSCAT(), ZCMAX, ALTMAX)
   FOR COUNT = 1 TO ZCMAX
      ALT2(COUNT) = ALT(COUNT)
   NEXT COUNT
   CALL SCNDSCATTER(GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), EPRIME(),
    EBRACKET(), ALT(), FIRSTSCAT(), SCNDSCAT(), ZCMAX, ALT2())
   CALL THIRDSCATTER(GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), EPRIME(),
    EBRACKET(), ALT(), FIRSTSCAT(), SCNDSCAT(), THIRDSCAT(), ZCMAX, ALT2())
   '*** OUTPUT FILE ALTITUDE VS GROUP GAMMA RATE ***
```

```
    OPEN "C:\THESIS\U8DATA\DOSE.5Y" FOR OUTPUT AS #2
    '************************************************
 CALL DOSECALC(GpS(), GT, MUt(), MUa(), ENERGY(), ACTIVITY, ALT(), FIRST-
SCAT(),
   SCNDSCAT(), THIRDSCAT(), ZCMAX, AREACONVERSION, MCiACTIVITY,
UNITLENGTH$,
   SCATCONT, SCNDSCATCONT, THIRDSCATCONT)
    TOTALDOSE = SCC + SCATCONT + SCNDSCATCONT + THIRDSCATCONT
    PRINT #2, "THE DOSE DUE TO DIRECT UNCOLLIDED PHOTONS IS ";
    PRINT #2, SCC; "R/HR"
    PRINT "THE TOTAL DOSE DUE TO DIRECT, FIRST, SECOND AND THIRD
SCATTER IS"
    PRINT TOTALDOSE; "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"
    PRINT #2, "THE TOTAL DOSE DUE TO DIRECT, FIRST, SECOND AND THIRD
SCATTER IS"
    PRINT #2, TOTALDOSE; "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$;
"^2)"
    CLOSE
    END


'********************************* SUBROUTINES
*****************************
    SETUP:
'*********************************************************************
********
'**  This subroutines initializes the unit area, total activity, and
local
'**  fraction for fallout.
'*********************************************************************
********
    INITIALIZE:
    CLS
    PRINT " ENTER THE UNIT OF LENGTH WITH WHICH TO CALCULATE"
    INPUT " THE UNIT AREA; MI, M OR KM (MI DEFAULT) ", UNITLENGTH$
    IF UNITLENGTH$ = "" THEN UNITLENGTH$ = "MI"
    UNITLENGTH$ = UCASE$(UNITLENGTH$)
    INPUT " ENTER THE ACTIVITY / UNIT AREA (MCi) (1MCi DEFAULT) ",
MCiACTIVITY
    IF MCiACTIVITY = Ø THEN MCiACTIVITY = 1
    PRINT " ENTER THE FRACTION OF THE ACTIVITY WHICH YOU FEEL WILL
FALL IN THE"
    INPUT " LOCAL AREA (1.Ø DEFAULT) ", LOCALFRACTION
    IF LOCALFRACTION = Ø THEN LOCALFRACTION = 1
    CLS
    PRINT " THE UNIT AREA IS "; UNITLENGTH$; "^2"
    PRINT " THE ACTIVITY DEPOSITED IS "; MCiACTIVITY; " MCi"
    PRINT " THE LOCAL FRACTION OF THE FALLOUT IS "; LOCALFRACTION
    INPUT " ARE THESE VALUES CORRECT (Y OR N)"; YORN$
    YORN$ = UCASE$(YORN$)
    IF YORN$ = "N" THEN GOTO INITIALIZE
    ACTIVITY = MCiACTIVITY * 1000000!  'CONVERT TO CURIES
```

```
    IF UNITLENGTH$ = "MI" THEN
      AREACONVERSION = 2560000
    ELSEIF UNITLENGTH$ = "KM" THEN
      AREACONVERSION = 1000000
    ELSEIF UNITLENGTH$ = "M" THEN
      AREACONVERSION = 1
    END IF
    RETURN

'*************************************************************************
********
    SUB DOSECALC (GpS(), GT, MUt(), MUa(), ENERGY(), ACTIVITY, ALT(),
     FIRSTSCAT(), SNDSCAT(), THIRDSCAT(), ZCMAX, AREACONVERSION, MCiAC-
TIVITY,
     UNITLENGTH$, SCATCONT, SCNDSCATCONT, THIRDSCATCONT)
'*************************************************************************
********
    CLS
    DIM SUB1DOSE(50), SUB2DOSE(50), SUB3DOSE(50)
    DIM SCAT1DOSE(50), SCAT2DOSE(50), SCAT3DOSE(50), SCNDDOSE(50)
    RNTGEN = .00877                    'UNIT CONVERSION ROENTGENS/(JOU-
LE/KG)
    JTOMEV = 1.602E-13                 'UNIT CONVERSION JOULE/MeV
    STOHR = 3600                       'UNIT CONVERSION SECONDS/HR
    SCATCONT = 0
    '*** FIRST CALCULATE FIRST SCATTER CONTRIBUTIONS
      FOR ZINC = 1 TO ZCMAX
      SUB1DOSE(ZINC) = 0
       FOR I = 1 TO 19
         X = ABS(ALT(ZINC) - 1) * MUt(I) * 1.225  'ABS(ALT(ZINC)-1) IS THE
           ' DIFFERENCE BETWEEN THE PLANE OF INTEREST AND THE
DETECTOR
           CALL EXPOFNCT(X, E1)
           SCAT1DOSE(I) = E1 * FIRSTSCAT(ZINC, I) * MUa(I) * .5 *
ENERGY(I)
           SUB1DOSE(ZINC) = SUB1DOSE(ZINC) + SCAT1DOSE(I)
        NEXT I
        IF ALT(ZINC) < 1 THEN
          DELZ = .1
        ELSEIF ALT(ZINC) < 10 THEN
          DELZ = 1
        ELSEIF ALT(ZINC) < 50 THEN
          DELZ = 5
        ELSE
          DELZ = 10
        END IF
      SCATCONT = SCATCONT + SUB1DOSE(ZINC) * DELZ
     PRINT ALT(ZINC); SCATCONT
     PRINT #2, ALT(ZINC); SCATCONT
    NEXT ZINC
    SCATCONT = SCATCONT * STOHR * JTOMEV / (AREACONVERSION *
```

```
RNTGEN)
   PRINT "TOTAL FIRST SCATTER CONTRIBUTION ="; SCATCONT;
   PRINT "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"
   PRINT #2, "TOTAL FIRST SCATTER CONTRIBUTION = "; SCATCONT;
   PRINT #2, "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"

   '*** CALCULATE SECOND SCATTER CONTRIBUTIONS
   '*** THESE CALCULATIONS ARE FOR THE PLANES OF ACTIVITY ABOVE
AND
   '*** BELOW THE DETECTOR
   SCNDSCATCONT = Ø
   FOR ZINC = 1 TO ZCMAX
     SUB2DOSE(ZINC) = Ø
      FOR I = 1 TO 19
        X = ABS(ALT(ZINC) - 1) * MUt(I) * 1.225  '*** ABS(ALT(ZINC)-1) IS
        ' THE DIFFERENCE BETWEEN THE PLANE OF INTEREST AND THE
DETECTOR ***
         CALL EXPOFNCT(X, E1)
       SCAT2DOSE(I) = E1 * SCNDSCAT(ZINC, I) * MUa(I) * .5 * ENERGY(I)
      SUB2DOSE(ZINC) = SUB2DOSE(ZINC) + SCAT2DOSE(I)
      NEXT I
       IF ALT(ZINC) < 1 THEN
         DELZ = .1
       ELSEIF ALT(ZINC) < 1Ø THEN
         DELZ = 1
       ELSEIF ALT(ZINC) < 5Ø THEN
         DELZ = 5
       ELSE
         DELZ = 1Ø
       END IF
     SCNDSCATCONT = SCNDSCATCONT + SUB2DOSE(ZINC) * DELZ
    PRINT ALT(ZINC); SCNDSCATCONT
    PRINT #2, ALT(ZINC); SCNDSCATCONT
   NEXT ZINC
   SCNDSCATCONT = SCNDSCATCONT * STOHR * JTOMEV / (AREACONVERSION
* RNTGEN)
   PRINT "TOTAL SECOND SCATTER CONTRIBUTION ="; SCNDSCATCONT;
   PRINT "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"
   PRINT #2, "TOTAL SECOND SCATTER CONTRIBUTION = "; SCNDSCATCONT;
   PRINT #2, "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"

   '*** CALCULATE THIRD SCATTER CONTRIBUTIONS
   '*** THESE CALCULATIONS ARE FOR THE PLANES OF ACTIVITY ABOVE
AND
   '*** BELOW THE DETECTOR
   THIRDSCATCONT = Ø
   FOR ZINC = 1 TO ZCMAX
     SUB3DOSE(ZINC) = Ø
      FOR I = 1 TO 19
        X = ABS(ALT(ZINC) - 1) * MUt(I) * 1.225  '*** ABS(ALT(ZINC)-1) IS
        ' THE DIFFERENCE BETWEEN THE PLANS OF INTEREST AND THE
```

```
DETECTOR ***
          CALL EXPOFNCT(X, E1)
        SCAT3DOSE(I) = E1 * THIRDSCAT(ZINC, I) * MUa(I) * .5 * ENERGY(I)
       SUB3DOSE(ZINC) = SUB3DOSE(ZINC) + SCAT3DOSE(I)
      NEXT I
       IF ALT(ZINC) < 1 THEN
         DELZ = .1
       ELSEIF ALT(ZINC) < 10 THEN
         DELZ = 1
       ELSEIF ALT(ZINC) < 50 THEN
         DELZ = 5
       ELSE
         DELZ = 10
       END IF
     THIRDSCATCONT = THIRDSCATCONT + SUB3DOSE(ZINC) * DELZ
    PRINT ALT(ZINC); THIRDSCATCONT
    PRINT #2, ALT(ZINC); THIRDSCATCONT
   NEXT ZINC
   THIRDSCATCONT = THIRDSCATCONT * STOHR * JTOMEV / (AREACONVER-
SION * RNTGEN)
   PRINT "TOTAL THIRD SCATTER CONTRIBUTION ="; THIRDSCATCONT;
   PRINT "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"
   PRINT #2, "TOTAL THIRD SCATTER CONTRIBUTION = "; THIRDSCATCONT;
   PRINT #2, "(R/HR)/("; MCiACTIVITY; "MCi/"; UNITLENGTH$; "^2)"
   END SUB


   SUB EXPOFNCT (X, E1)
'*************************************************************************
********
'*  This routine calculates the approximation for the EXPONENTIAL
*
'*  FUNCTION comes from Abramawitz and Stegan.
*
'*************************************************************************
********
    IF X < 1 THEN                    ' X=MUt*S*1.225, MUt=MUt/RHO
      A0 = -.57722566#
      A1 = .99999193#
      A2 = -.24991055#
      A3 = .05519968#
      A4 = -.0097604
      A5 = .00107857#
       E1 = A0 + A1 * X + A2 * X ^ 2 + A3 * X ^ 3 + A4 * X ^ 4 +
A5 * X ^ 5
       E1 = E1 - LOG(X)
'       PRINT E1, X
    ELSE
      A1 = 2.334733
      A2 = .250621
      B1 = 3.330657
      B2 = 1.681534
```

```
        E1 = X ^ 2 + A1 * X + A2
        E1 = E1 / (X ^ 2 + B1 * X + B2)
        E1 = E1 / (X * EXP(X))
        PRINT E1
      END IF
    END SUB

    SUB FIRSTFLIGHT (GpS(), GT, MUt(), MUa(), ENERGY(), ACTIVITY, LOCAL-
FRACTION,
     MCiACTIVITY, AREACONVERSION, UNITLENGTH$, SCC)
'*************************************************************************
********
'** This module will calculate the dose at one meter above an infinite
**
'** homogeneous plane of activity for the first flight (uncollided)
**
'** photons.
**
'*************************************************************************
********
     SUMTOT = 0
     FOR I = 1 TO 19
        FRACTION = GpS(I) / GT
          X = MUt(I) * 1.225
         CALL EXPOFNCT(X, E1)
          PRINT USING " #.###^^^^  "; ENERGY(I); GpS(I); FRACTION; E1
         PART = FRACTION * MUa(I) * E1 * ENERGY(I)
        SUMTOT = SUMTOT + PART
     NEXT I
     SCC = 1216.425 * SUMTOT * ACTIVITY * LOCALFRACTION / AREACON-
VERSION
    PRINT ""
    PRINT "  SOURCE CONVERSION COEFFICIENT, SCC = ";
    PRINT USING "#.####^^^^"; SCC;
    PRINT " (R/HR)/("; MCiACTIVITY * LOCALFRACTION; "MCi/"; UNITLENGTH$;
"^2)"
    END SUB

    SUB FIRSTSCATTER (GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), ACTIVITY,
     LOCALFRACTION, EPRIME(), EBRACKET(), ALT(), FIRSTSCAT(), ZCMAX, ALT-
MAX)
'*************************************************************************
********
'*  This program will calculate step 1) described below to begin the
*
'*  determination of the the gamma ray dose contribution due to
*
'*  scattered photons.  The program employs the method of successive
*
'*  scatters.  See Bridgeman's notes Chapter 5.
```

```
*
'*
*
'*  Step 1)  Calculate the scattering rate at all altitudes to 150
*
'*  meters.  The following altitudes were used: 0.05, 0.15, 0.25, 0.35, *
'*  0.45, 0.55, 0.65, 0.75, 0.85, 0.95, 1.5, 2.5, 3.5, 4.5, 5.5, 6.5,   *
'*  7.5, 8.5, 9.5, 12.5, 17.5, 22.5, 27.5, 32.5, 37.5, 42.5, 47.5, 55,  *
'*  65, 75, 85, 95, 105, 115, 125, 135, 145.  These values were used as
*
'*  cell centers, provides accurate results and facilitate the
*
'*  integration in Step 2).  The scattering rate at these altitudes
can *
'*  be described as a volume distributed source of photons whose
energy *
'*  is less than the initial photon.
*
'*
*
'*  Step 2)  Determine the contribution of this volume distributed
*
'*  source by integrating over all volume.
*
'*
*
'*  Step 3)  This method can be expanded to include 2nd, 3rd, etc
*
'*  scatters
*
'*****************************************************************
********
   CLS
   DIM SCATCONT(50, 20)
   C = 3.7E+10              ' DISINTEGRATIONS PER SECOND PER CURIE OF
ACTIVITY
   AC = ACTIVITY * C * LOCALFRACTION
   'FOR ZC = 1 TO 50
   '   FOR I = 1 TO 19
   '     SCATTOT(ZC, I) = 0
   '   NEXT I
   'NEXT ZC

'********** CALCULATE THE SCATTERING RATE AT VARIOUS ALTITUDES
**********
'*************** ASSUME NO DENSITY CHANGES WITH ALTITUDE
***************
    ZC = 1                              'INCREMENTER
     ALT(ZC) = .05                     'INITIAL ALTITUDE
    DO WHILE ALT(ZC) <= ALTMAX
     PRINT ALT(ZC)
```

```
      FOR I = 1 TO 19
       X = ALT(ZC) * MUt(I) * 1.225
        CALL EXPOFNCT(X, E1)
'************ THE SCATCONT(ALT,ENERGY)  = FLUX*CROSS SECTION
************
       FRACTION = GpS(I) / GT                  'FRACTION OF TOTAL
ACTIVITY WHICH
                                                'CONTRIBUTES TO THE
Ith ENERGY GRP
         FLUX = AC * FRACTION * (.5 * E1)      'SOURCE ATTENUATED TO
ALTITUDE
         SCATCONT(ZC, I) = FLUX * (MUs(I) * 1.225) 'SCATTERING CONTRI-
BUTION
                                                   'FROM THE Ith
ENERGY GRP
   ENERGYBIN:
' THIS ROUTINE DETERMINES WHICH ENERGY BIN TO ASSIGN THE SCATTERED
PHOTON
        FOR J = 1 TO 20
         IF EPRIME(I) < EBRACKET(J) THEN
           FIRSTSCAT(ZC, J - 1) = FIRSTSCAT(ZC, J - 1) + SCATCONT(ZC,
I)

           GOTO GOON
         END IF
        NEXT J
   GOON:
'*****************************************************************
********
      NEXT I                                    ' NEXT ENERGY
       IF ALT(ZC) < .9 THEN                    'SERIES OF IF STATEMENTS
         ALT(ZC + 1) = ALT(ZC) + .1   'DETERMINE HOW THE ALTITUDE
       ELSEIF ALT(ZC) < 1 THEN                 'INCREMENTS TO THE NEXT
STEP.
         ALT(ZC + 1) = ALT(ZC) + .55
       ELSEIF ALT(ZC) < 9 THEN
         ALT(ZC + 1) = ALT(ZC) + 1
       ELSEIF ALT(ZC) < 10 THEN
         ALT(ZC + 1) = ALT(ZC) + 3
       ELSEIF ALT(ZC) < 45 THEN
         ALT(ZC + 1) = ALT(ZC) + 5
       ELSEIF ALT(ZC) < 50 THEN
         ALT(ZC + 1) = ALT(ZC) + 7.5
       ELSE
         ALT(ZC + 1) = ALT(ZC) + 10
       END IF
      ZC = ZC + 1
    LOOP
    ZCMAX = ZC - 1
END SUB
```

```
'****************************** SUBROUTINES
*******************************
   SUB SCNDSCATTER (GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), EPRIME(),
   EBRACKET(),ALT(), FIRSTSCAT(), SCNDSCAT(), ZCMAX, ALT2()  '***********
*************************************************************
'*
*
'*  This program will calculate the scattering rate of the second
*
'*  scattered photons.
*
'*
*
'*************************************************************************
********
   CLS
   DIM SCATRATE(5Ø, 19)
'**************  THESE CALCULATIONS ARE FOR THE PLANES OF
*************
'**************  ACTIVITY ABOVE AND BELOW THE SECOND PLANE
*************
  FOR ZINC = 1 TO ZCMAX              '*** EACH FIRST SCATTER PLANE
(SOURCE)
   PRINT ALT(ZINC)
    FOR ZC2 = 1 TO ZCMAX             '****** EACH PLANE OF SECOND
SCATTER
     PRINT ALT2(ZC2)
       FOR I = 1 TO 19                 '*************** EACH ENERGY
        SCATRATE(ZC2, I) = Ø
        X = ABS(ALT(ZINC) - ALT2(ZC2)) * MUt(I) * 1.225
'******** ABS(ALT(ZINC)-ALT2(ZC2)) IS THE DIFFERENCE BETWEEN THE
********
'***  PLANE OF FIRST SCATTERING AND THE PLANE OF SECOND SCATTERING
***
          IF X = Ø THEN
             SCATRATE(ZC2, I) = Ø      '*** EXPONENTIAL INTEGRAL
UNDEFINED
             GOTO SAMEPLANE             '*** FOR X=Ø, SCATTER ON
SAME PLANE
          END IF
          CALL EXPOFNCT(X, E1)
'********** SCATTERING RATE IS EQUAL TO THE FLUX * CROSS SECTION
*********
           IF ALT(ZINC) < 1 THEN
             DELZ = .1
           ELSEIF ALT(ZINC) < 1Ø THEN
             DELZ = 1
           ELSEIF ALT(ZINC) < 5Ø THEN
              DELZ = 5
           ELSE
              DELZ = 1Ø
```

```
            END IF
            FLUX = FIRSTSCAT(ZINC, I) * (.5 * E1) * DELZ
   '******  SOURCE ATTENUATED TO ALTITUDE OF REACTION  ******
            SCATRATE(ZC2, I) = FLUX * (1.225 * MUs(I))
   SAMEPLANE:
'******** DETERMINE WHICH BIN TO ASSIGN THE SCATTERED PHOTON INTO
*******
              FOR J = 1 TO 2Ø
               IF EPRIME(I) < EBRACKET(J) THEN
                SCNDSCAT(ZC2, J - 1) = SCNDSCAT(ZC2, J - 1) +
SCATRATE(ZC2, I)

                GOTO GOON2
               END IF
              NEXT J
   GOON2:
       NEXT I          ' NEXT ENERGY
     NEXT ZC2        ' NEXT SCATTERING (REACTION) ALTITUDE
   NEXT ZINC      ' NEXT SOURCE ALTITUDE
   END SUB


   SUB THIRDSCATTER (GpS(), GT, MUt(), MUa(), MUs(), ENERGY(), EPRIME(),
    EBRACKET(), ALT(), FIRSTSCAT(), SCNDSCAT(), THIRDSCAT(), ZCMAX, ALT2())
'*************************************************************
********
'*
*
'*   This program will calculate the scattering rate of the third
*
'*   scattered photons.
*
'*
*
'*************************************************************
********
   CLS
   DIM SCATRATE3(5Ø, 19)
'*************  THESE CALCULATIONS ARE FOR THE PLANES OF
*************
'*************  ACTIVITY ABOVE AND BELOW THE SECOND PLANE
*************
    FOR ZINC = 1 TO ZCMAX            '*** EACH FIRST SCATTER PLANE
(SOURCE)
     PRINT ALT(ZINC)
      FOR ZC2 = 1 TO ZCMAX           '****** EACH PLANE OF SECOND
SCATTER
       PRINT ALT2(ZC2)
       FOR I = 1 TO 19               '************** EACH ENERGY
        X = ABS(ALT(ZINC) - ALT2(ZC2)) * MUt(I) * 1.225
'****** ABS(ALT(ZINC)-ALT2(ZC2)) IS THE DIFFERENCE BETWEEN THE
*********
```

```
'****  PLANE OF FIRST SCATTERING AND THE PLANE OF SECOND SCATTER-
ING  ***
          IF X = Ø THEN
             SCATRATE3(ZC2, I) = Ø '*** EXPONENTIAL INTEGRAL
UNDEFINED
             GOTO SAMEPLANE2          '*** FOR X=Ø, SCATTER ON
SAME PLANE
          END IF
          CALL EXPOFNCT(X, E1)
'********** SCATTERING RATE IS EQUAL TO THE FLUX * CROSS SECTION
********
          IF ALT(ZINC) < 1 THEN
            DELZ = .1
          ELSEIF ALT(ZINC) < 1Ø THEN
            DELZ = 1
          ELSEIF ALT(ZINC) < 5Ø THEN
            DELZ = 5
          ELSE
            DELZ = 1Ø
          END IF
           FLUX = SCNDSCAT(ZINC, I) * (.5 * E1) * DELZ
   '******  SOURCE ATTENUATED TO ALTITUDE OF REACTION ******
           SCATRATE3(ZC2, I) = FLUX * (1.225 * MUs(I))
   SAMEPLANE2:
'******** DETERMINE WHICH BIN TO ASSIGN THE SCATTERED PHOTON INTO
*******
        FOR J = 1 TO 2Ø
          IF EPRIME(I) < EBRACKET(J) THEN
           THIRDSCAT(ZC2, J - 1) = THIRDSCAT(ZC2, J - 1) + SCATRA-
TE3(ZC2; I)
           GOTO GOON3
          END IF
        NEXT J
'*****************************************************************
********
   GOON3:
       NEXT I          ' NEXT ENERGY
     NEXT ZC2        ' NEXT SCATTERING (REACTION) ALTITUDE
   NEXT ZINC      ' NEXT SOURCE ALTITUDE
   END SUB
```

Appendix J

Description of DKPOWR

DKPOWR was developed in conjunction with the American National Standards Institute/American Nuclear Society ANSI/ANS-5.1-1979 Standard for Decay Heat Power in Light Water Reactors. The code uses nonlinear least squares fitting of a series of exponential functions to data from CINDER-10 and experimental observations. CINDER-10 calculations were based on nuclear data from Version IV of the Evaluated Nuclear Data File ENDF/B-IV. (16:1.1)

Experimental observation has shown that isotope generation and depletion codes based on commonly used nuclear data tables, i.e. ENDF/B-IV, are deficient in early time high energy gamma-rays, resulting in calculated spectra which are softer than the actual spectra. The gamma-ray energy spectra for times < $1X10^{05}$ seconds were supplemented by experimental results for U-235 and Pu-239 (16:3.9-3.10), however the major discrepancies between experimental and calculated values occurs at times of less than approximately 20 minutes. No experimental data is yet available for U-238 (6:*).

The input file for DKPOWR is very simple and requires a minimum of data. A brief description of each line of a sample input file shown in the following table.

Table 1 Description of DKPOWR Input File

Record	Column	Description
Title	1-8Ø	Problem Definition (Comment Line)
Control	1-4	NNUC Number of fuels If NNUC < Ø input average fission rate If NNUC > Ø input total number of fissions
	5-8	NTPOW Number of pulses If NTPOW < Ø input time in hours If NTPOW > Ø input time in seconds
	9-12	NTCOOL Number of cooling times If NTCOOL < Ø input time in hours If NTCOOL > Ø input time in seconds
	13-16	JTUNIT Output time units 1=Seconds, 2=Hours, 3=Days
	17-2Ø	JPUNIT Output decay power and spectra units
	21-24	1=MeV/S, 2=Watts JEUNIT Output decay energy units 1=MeV, 2=Joules, 3=Watt-Hours
Uncertainty	1-48	Used with ANSI/ANS-5.1-1979 Standard
Fission History	(6e15.5)	Fission pulse time increments
Cooling Times	(6e12.5)	Cooling times
Power Step ID	1-4	Pulse number
Fission Level	1-6	Nuclide ID=Z*1ØØØØ+A*1Ø+S S=State Ø=Ground, 1=1st Isomeric, etc
	7-18	FRATE Average fission rate or total fissions
	19-3Ø	Average capture rate or total number of captures for u-238

The output from DKPOWR is very straightforward. The input information is echoed in tables at the front of the output file, while each output result is clearly labeled in individual tables. The spectra are presented in separate tables at each cooling time. Table 2 shows the energy bin structure used for the gamma ray spectra. More detailed descriptions of the program and associated files are contained in LA-UR-85-157 (16:2.5-4.3).

DKPOWR is available at AFIT from Charles J. Bridgman of the Engineering Physics Dept.. Once the input file is created, execute the code by typing:
DKPOWR.OUT
To run the code in the background type:
nohup DKPOWR.OUT &

Table 2 DKPOWR Gamma Ray Spectra Energy Bin Structure

Bin #	Lower Bound (MeV)	Midpoint (MeV)	Upper Bound (MeV)
1	Ø.Ø	Ø.Ø5	Ø.1
2	Ø.1	Ø.15	Ø.2
3	Ø.2	Ø.3	Ø.4
4	Ø.4	Ø.5	Ø.6
5	Ø.6	Ø.7	Ø.8
6	Ø.8	Ø.9	1.Ø
7	1.Ø	1.1	1.2
8	1.2	1.3	1.4
9	1.4	1.5	1.6
1Ø	1.6	1.7	1.8
11	1.8	1.9	2.Ø
12	2.Ø	2.1	2.2
13	2.2	2.3	2.4
14	2.4	2.5	2.6
15	2.6	2.8	3.Ø
16	3.Ø	3.5	4.Ø
17	4.Ø	4.5	5.Ø
18	5.Ø	5.5	6.Ø
19	6.Ø	6.75	7.5

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Vita

Kyle K. Millage was born on 13 July 1962 in Elwood, Indiana. He graduated from Shenandoah High School, Middletown, Indiana in 1980, and attended Purdue University, receiving his Bachelor of Science degree in Nuclear Engineering upon graduation in 1984. He was commissioned in the US Air Force in August 1984, and was assigned to the Technical Operations Division/AFTAC at McClellan AFB, California. Captain Millage entered the nuclear engineering program at the Air Force Institute of Technology in September 1987.

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The purposes of this study were to determine fission product decay characteristics, including the total activity, the gamma-ray emission rate (GER) and gamma-ray energy spectra. The activity and GER decay were compared to Way and Wigner's $t^{-1.2}$ approximation, and the effects the spectra, activity and GER have on the Source Normalization Constant (K) were examined. Most of the fission product data were obtained from DKPOWR, and were compared with data obtained from ORIGIN2. Since the gamma rays are of primary concern in fallout studies, the GER is used instead of activity. The ratio of GER to activity changes significantly with time. The results of this study calculate a GER of 590×10^{16} gamma rays/second per kT of fission yield from U-235 fuel and a K of 7059 R/Hr/(kT/km^2). The calculation of K includes the contribution from scattered photons. The GER result is 11% higher than reference values, while the K is within 2% of the current value in Glasstone and Dolan's The Effects of Nuclear Weapons. The Ks for Pu-239 and U-238 were within 5% of the U-235 results. The Way-Wigner $t^{-1.2}$ approximation differs from time dependent GER and K up to 85% for times less than 6 months. The approximation is not valid for the GER or K at times greater than 6 months. The approximatin is within about 45%, for the activity from fission product decay to atleast 5 years. A more accurate measure of exposure requires a numerical integration of the time dependent GER and Source Normalization "Constant".